ThorCon Breeder Reactor Design

MOLTEN SALT THORIUM REACTOR

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SECTION 1- Executive Summary

To date, molten salt reactors (MSRs) are the only liquid fuel reactors that are expected to undergo commercialization in the 21st century, due to their safe and reliable design theories. Today, research is still being conducted in MSRs and their capabilities to produce electricity at a larger scale and for a longer time frame. Companies such as ThorCon Power are designing walk away safe, reliable, and relatively small liquid fueled reactors. ThorCon plans to use the idea of a shipyard construction to mass produce safe, inexpensive power plants that can provide electricity all over the globe.

In working with ThorCon Power and MSR was designed in hopes to help meet the demand of Indonesia in its effort to implement 5 GW of "new power". The goal of the designed MSR in this report is to be able to provide at least 20% of this power with an innovated ThorCon Power reactor design.

The MSR designed in this report incorporates many features of the original ThorCon Power reactor and implements aspects of an SM-AHTR (Small Modular Advanced High Temperature Reactor) reactor for convenience and ease of assembly. The goal of the MSR in this report, the ThorCon Breeder, is to produce a reactor that could provide 250 MW of electricity over the course of four years and a shutdown and maintenance time frame of one year. The ThorCon Breeder nuclear plant will operate on 4 cores to produce a total of 1 GW of electricity, with an empty ThorCon Breeder can on standby.

The ThorCon Breeder was designed starting with the basic geometry of the core based on its buckling calculations. Once the size and geometry of the core was finalized the power cycle of the reactor and its neutronics were worked on simultaneously for iteration purposes. The fuelsalt of the ThorCon Breeder was derived from ThorCon original reactor design but innovated to produce a breeding reactor. The fuelsalt was divided in to a breeding salt and a fuelsalt region which plays a key role in the geometry of this reactor design.

The ThorCon Breeder implements the simple “can within a can” design of the Sm-AHTR by developing a homogenized pool of fuelsalt in the center of the core that is held by a graphite (moderating) cylinder that is surrounded by the thorium breeding salt which is encased in another graphite cylinder (reflector) which is all encased in a reactor vessel made of Hastelloy-N. In order to create breeding system research was conducted on the chemical processing and fuel cycle of the thorium salt which is converted in to the uranium fuelsalt throughout out the reactors operation, this serves as a key element of this design.

The power cycle chosen for this design was an open-air Brayton cycle due to the ease o obtaining air and releasing the waste air to run a bottoming Rankine cycle. This power cycle provides a total of 329 MW of electricity from each reactor giving a total of 1.32 GW of electricity with an efficiency of 52%, which exceeds the 20% of electricity hoped to achieve for Indonesia. This power cycle operates on a single loop between the air and fuelsalt of the reactor.

Neutronic calculations enabled the determination of the neutron flux, start up time, burn up of the fuel, and the final geometry of the reactor. The reactor is 5.96 m high and 5.8 m in width which is very near the size of an original ThorCon reactor.

The ThorCon Breeder reactor is designed to operate with safety designs that enable it to be a reactor that is essentially “walk-away” safe. This means that in case of the worst accidents the reactor can shut itself down without human intervention. This report also covers the cost of the plant which is marginally near the cost of the original ThorCon reactor.

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SECTION 2-Introduction

HISTORY

The premise for molten salt reactors (MSRs) originated from the Nuclear Energy for the Propulsion of Aircraft (NEPA) program, which was initiated by the Unites States Army Air Forces in 1946. The NEPA program was eventually transformed into the Aircraft Nuclear Propulsion (ANP) program, but the goal of the program remained unchanged: to develop a reliable method of nuclear propulsion for United States aircraft. The NEPA/ANP design would be a 2.5 megawatt thermal nuclear reactor engine for a bomber aircraft that ran approximately 100 hours. NEPA's continued work led to experimentation in graphite-moderated molten salt reactors. These experiments included improvements of fuel interactions, the control of reactivity, and nuclear stability. Between the 1940s and 1950s, Oak Ridge discovered that thermal reactors operating on a thorium fuel cycle would be the most feasible molten-salt system for the production of economic power, providing passive safety without the concern of the core melting, low pressure in the primary cooling loop, and the potential for operation in both the thermal and fast neutron spectrum. To date, the molten salt reactor is the only liquid fuel reactor that is expected to undergo commercialization in the 21st century[22].

Today, research is still being conducted in MSRs and their capabilities to produce electricity at a larger scale and for a longer time frame. Companies such as ThorCon Power are designing walk away safe, reliable, and relatively small liquid fueled reactors. ThorCon plans to use the idea of a shipyard construction to mass produce safe, inexpensive power plants that can provide electricity all over the globe. ThorCon requires unique designs that meet the power requirements of their consumers, uses liquid fuel and coolant, and remain cost effective.

PROBLEM STATEMENT

Access to energy is required for economic growth. Yet 1.5 billion people in the world today do not have access to electricity[3]. Most of this population resides in developing countries, and as they continue to grow, their demand for energy will rise. Providing affordable electricity for developing countries, such as Indonesia, is a problem that we believe can be solved with the implementation of small molten salt reactor power plants. MSRs can provide power on a large scale while reducing the dependence on fossil fuels. These advantages eliminates political debate over fuel price and reduces the strain of energy costs on economic growth. MSR Company, ThorCon Power, has requested that we design a nuclear power plant that meets the requirements of the aforementioned consumer while remaining affordable. We plan to design a nuclear power plant using ThorCon's liquid fuel salt mixture, which can provide 1 GWe of power to Indonesia or any developing country

ThorCon has stipulated that our MSR design analyze two cases:

Case 1: The neutronic reactivity feedback that would result from the leak of the thorium breeding material in a pipe rupture or malfunction in chemical processing.

Case 2: Conduct a safety analysis of the reactor in an emergency core drain scenario. Analyzing the time frame of the freeze valve system response.

Both cases will be analyzed numerically, using MCNP 6, and analytically, using thermodynamic and hydraulic concepts.

MSR METHODOLOGY

Liquid-fuel MSRs can be designed for either homogeneous or heterogeneous cores. In homogeneous designs thorium can be dissolved with uranium in a single fluid to serve as the fuel, while two-fluid or heterogeneous designs operate as a breeding reactor. In this design, a fertile thorium salt is separated from the fuel salt that contains fissile uranium or plutonium. The fertile thorium salt is placed around the core of the uranium fuel salt so that it can breed more fissile material into the fuel.

In a standard molten salt reactor concept, the fuel is a molten mixture of lithium and beryllium fluoride salts (known colloquially as FLiBe). Low-enriched uranium (U235 or U233) fluoride, UF_4, is added to the aqueous FLiBe solution. The core consists of a graphite moderator that permits the salt to flow at a low pressure (~ 1 atm) and a temperature of about 700^{o}C. The basic design for a MSR uses epithermal neutrons that are moderated by the graphite. In basic design operations, heat is transferred to a secondary salt circuit and then to a steam or process-heat generator to produce electricity.

ADVANTAGES

A thorium liquid fueled MSR provides many advantages over more traditional light water reactor (LWR) designs, the most important being heightened safety. Our MSR design circumvents the problems associated with a meltdown scenario. Using salt as a coolant allows the reactor vessel to be at atmospheric pressure, so the pressure vessel is easier to construct and safer to operate. The salt coolant also enables the reactor to be run at a higher temperature, which can lead to greater efficiency through use of a Brayton combined cycle. Using thorium as the fuel breeder makes the MSR proliferation resistant because thorium is not a fissile isotope. The natural isotopic abundance of thorium is dominated by its fertile isotope Th-232, which is used in this reactor design to breed fissile U-233. In comparison to natural uranium, which is only about 0.7% U-235 (the fissile isotope), natural thorium is essentially pure Th-232, this means that fueling a thorium reactor would be several orders of magnitude cheaper because the cost of enrichment is negligible. This makes thorium cheaper on a per mass basis than uranium, which is a significant advantage in designing a nuclear reactor.

DISADVANTAGES

Though all of these advantages look attractive, one must still consider the challenges and disadvantages to a thorium liquid fueled MSR. The salt used in the reaction will produce tritium when bombarded with the neutrons native to a fission reactor. Tritium is highly dangerous if it enters the body because it has a long biological and effective half-life. Using the molten salt as a coolant will also require a multi-loop design because a salt turbine is not possible with current technology. Another issue is that an MSR has never been certified by the NRC due to the fact that the design is relatively new and research on their reliability and inherent safety features is still being conducted, so this may propose as a challenge in actual development.

APPLICATIONS

The MSR design could have some unique implications. For one, the modular design would be ideal for developing nations because modules are cheaper to operate and can be purchased in stages, whereas an AP 1000 or ABWR requires a massive outlay of money. Another unique function of this design is the high-temperature waste heat used in an industrial process[7].

REPORT OUTLINE

This design report will explain and analyze the theoretical model for a ThorCon Breeder reactor that is a homogenized, salt cooled, and liquid fueled. We will provide the design specifications of the nuclear plant and its start-up and operation cycle. The fuel, coolant, and moderator requirements will be discussed as it pertains to thermodynamics and the neutronics of the core. The core will be analyzed in terms of size and the materials used to construct the reactor vessel for safety and power production analysis. Components of the plant, such as the turbine and compressors will be analyzed for their efficiency.

This will be followed by the examination of ThorCon's specified cases and our nuclear power plant's ability to meet the required consumer needs. A thorough safety analysis of our overall design and power output and thermal efficiency will be conducted.

Conclusively, the report will analyze our design's complications and proposed solutions, along with alternative methods of operation and design features that could be implemented in the design of the nuclear power plant.

SECTION 3: Theoretical Model

3.1 ThorCon Power Literature Review

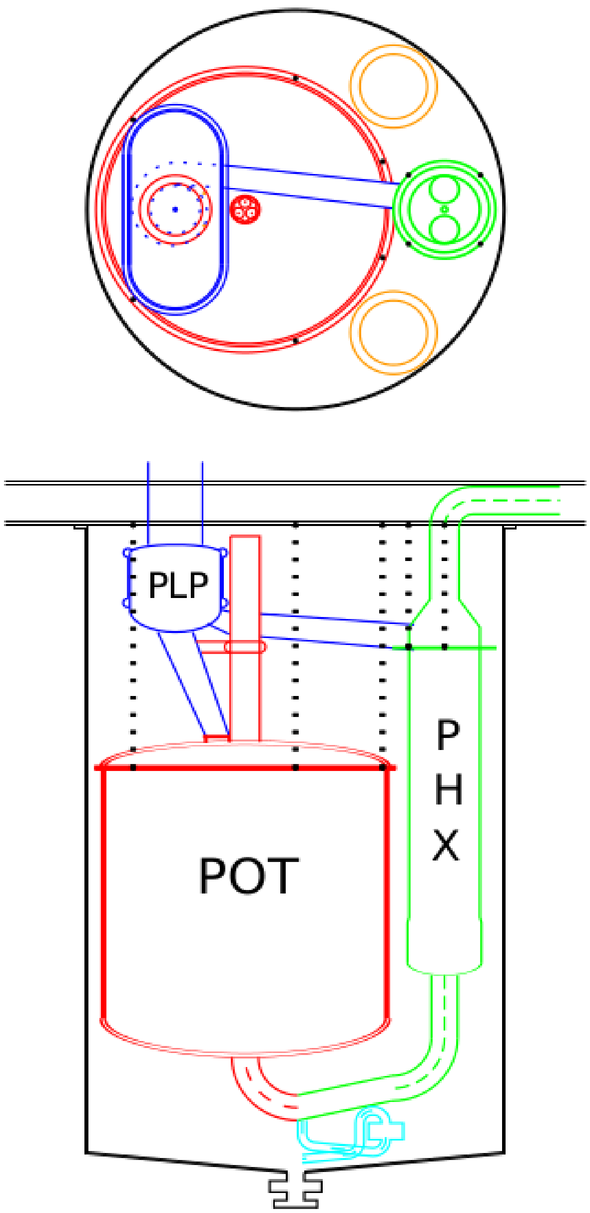
ThorCon is a simple molten salt reactor and differs from current reactors in that the fuel is in liquid form. ThorCon is designed so that if the reactor overheats for any reason, the reactor will automatically shut itself down and drain the fuel from the primary loop into a decay tank, thus passively handling the decay heat. ThorCon designs are such that the reactor is essentially "walk away safe," meaning that in case of overheating emergencies there is no need for operator intervention.

The ThorCon reactor is 30 m underground and operates at near-ambient pressure [7]. In the case of a primary loop rupture, there is no phase change in the fuel salt. The most troublesome fission products, including strontium-90 and cesium-137, are chemically bound to the salt and in fuel drain scenarios, will end up in the drain tank as well[4].

The entire ThorCon plant is manufactured in blocks on a shipyard-like assembly line. ThorCon has a systemized way of building power plants, such that a single large reactor shipyard can produce one hundred 1 GWe ThorCons per year. This manufacturing capability allows consumers to purchase modular components as needed.

Every component of the ThorCon reactor is replaceable with little or no interruption in power output and ThorCon reactors are designed such that all key components are replaced after a short period of time. For example, the graphite reflectors are replaced every four years.

ThorCon power plants require less resources than a coal plant, thus making it a cheaper source of energy. ThorCon can produce reliable, carbon-free, electricity at 3-5 cents per kWh, depending on the scale [4]. Figure 3.1-1 shows the design and parameters of a ThorCon.



Can electrical output(MW):	250.0
Can thermal output(MW):	557.0
Fuelsalt:	NaF-BeF2-ThF4-UF4
Mol percent:	76/12/9.8/2.2
Fuelsalt flow rate(kg/s):	2934
Pot temperature in(C):	564.0
Pot temperature out(C):	704.0
Loop transit time(seconds):	13.8
Pot inlet pressure(bar g):	1.65
Primary Loop Pump(kW):	1006
Net U fissile consumed(kg/y):	112
Overall plant efficiency(%):	44.9
Percent thorium fueled(%):	25.1
Can OD(m):	7.259
Can Height(m):	11.738
Pot OD(m):	4.961
Pot Height(m):	5.717
Can weight (no salt) (kg):	381,767
Fuelsalt weight (kg):	40,515
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Version:	1.08

Figure 3.1-1- ThorCon Power Reactor Design[6]

3.1.1 ThorCon Reactor Key Components

Our ThorCon Breeder reactor design implements 2 key components of the original ThorCon reactor. The membrane wall and the fuel salt drain tank are used in our design for their unique safety features.

3.1.1.1 Membrane Wall

The membrane wall is an important feature of a ThorCon reactor assembly. Each reactor vessel is encased in this membrane wall, as shown in Figure 3.1.1.1-1. The membrane wall has a cooling feature that is made of vertical steel tubes connected by strips of steel plates which are welded to the tubes [5]. These tubes are filled with water and are connected by headers at the top and bottom so that the water can circulate from the top to the bottom of the wall. The ThorCon reactor vessel is cooled by thermal radiation to the membrane wall. As the heat is released from the reactor, a portion of the water in the tubes is converted into steam a rises to the top by natural circulation to a cooling pond where the steam condenses and is then returned to the pipes from the bottom of the membrane wall.

The membrane wall cooling system will be used to keep our reactor vessel cool during operation, which will limit the thermal stress on the material used for our reactor vessel, Hastelloy-N. Our reactor will operate at 1000^{o}C at the core, which is about 300^{o} higher than that of the original ThorCon design. We predict that with the use of the membrane wall cooling system, this increase in temperatures will not significantly affect the reactor vessel.

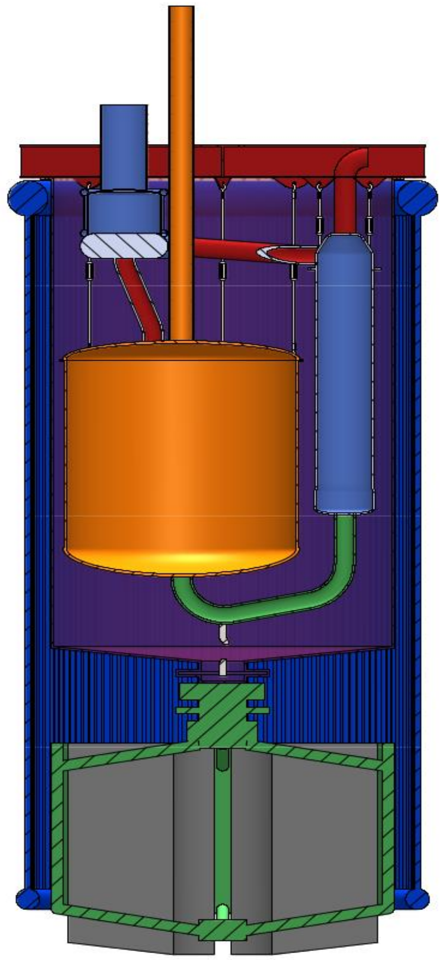

Figure 3.1.1.1-1: Membrane Wall Cooling System- Blue portion of the drawing. The wall serves as a cooling mechanism for the ThorCon Reactor Vessel and Fuelsalt drain tank. (Design by Nathan Crum of ThorCon Power)

3.1.1.2 Fuelsalt Drain Tank

The ThorCon Power fuelsalt drain tank sits directly below the reactor vessel. Shown in Figure 2 is the decay tank in green. At the bottom of the reactor vessel is a freeze valve that holds frozen fuelsalt, also shown in Figure 2 in gray, which serves as a plug between the reactor and the drain tank. This valve will remain frozen and melt in overheating cases, allowing the fuelsalt from the core to drain into the drain tank. Because there is no moderating material with in this tank fission will stop almost immediately [5].

The fuelsalt drain system is a passive operation, and there is nothing that an operator can do to prevent it. We plan to redesign the plug system so that the overheating parameter does not solely rely on the melting point of the frozen fuel salt (~350^{o}C). This way our operating temperatures of 1000^{o}C, will not affect the freeze plug at normal operation temperatures. The freeze plug innovation and analysis will be further explained in *Section 4.1.2 Freeze Valve Analysis.*

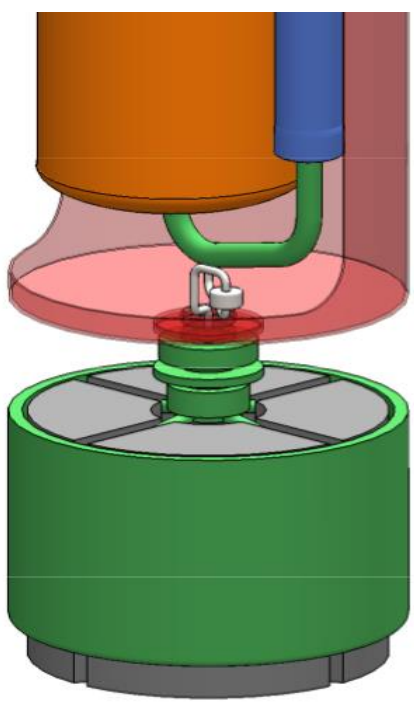

Figure 3.1.1.2-1: Fuelsalt Drain Tank- Green canister in the drawing. The overheated fuel inside of the yellow reactor vessel drains into the decay tank through the thawed freeze valve (gray tube portion). (Design by Nathan Crum of ThorCon Power)

3.2 Plant Design Specifications

3.2.1 Research

To begin the design process, we researched reactor designs the met our requirements for the project. This was done so that we could incorporate innovative aspects into our design and use the researched designs as a way of comparison as our own design progressed. Since MSR designs are very novel and not much research has been done in their operation and design. However, we are incorporating the Sm-AHTR (small modular advanced high temperature reactor) developed by ORNL (Oak Ridge National Lab) with the design aspects of the ThorCon Power Reactor. The Sm-AHTR meets our requirement of a small-modular design, easy component replacement, and inexpensive power conversion. The basic design specifications of the Sm-AHTR can be found in Table 3.2.1-1 and a drawing of its design in Figure 3.2.1-1.

Design Parameters	Value
Power (MWt/MWe)	125/50+
Primary Coolant	Li-BeF_2
Core Inlet Temperature (^oC)	650
Core Outlet Temperature (^oC)	700
Operational Heat Removal	3-50% loops
Passive Decay Heat Removal	3-50% loops
Power Conversion	Brayton

Reactor Vessel Penetrations	None

Table-3.2.1-1: Overall System Parameters of the Sm-AHTR (ORNL)

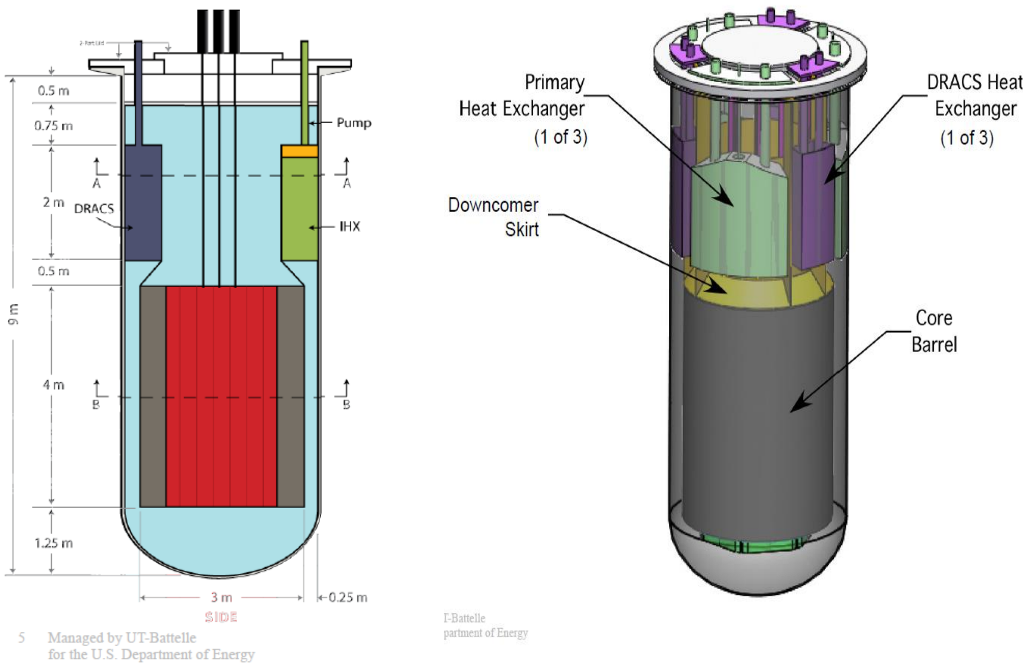

Figure-3.2.1-1: Sm-AHTR Design

The Sm-AHTR and the ThorCon are both molten-salt cooled reactors with removable core components that meet our need of replacing core parts every four years, due to degradation. Both cores operate near atmospheric pressure, which also aligns with our design desire. These design are also similar in that their heat exchanger systems are near or on top (Sm-AHTR) of the core so that the primary loop never leaves the reactor vessel. This design feature is a key requirement of our design as it pertains to the safety of our model.

The Sm-AHTR and ThorCon differ in that the Sm-AHTR is designed for solid fuel plates or rods, while ThorCon is operating with liquid fuel that is mixed with the coolant. We will stay true to ThorCon's design and use a liquid fuel. Our fuel will differ from ThorCon in that it will not be mixed with the material needed to breed the fuel, we will incorporate a mixture specifically for breeding and fuelsalt mixture (discussed in *Section 3.4.2 Fuel and Breeder Chemical Composition).*

Referring to *Section 3.1 ThorCon Power Literature Review*, our design will match the power output of 250 MWe and thus will be roughly the same size. The Sm-AHTR's power conversion use of a Brayton cycle and stackable reactor component feature will be incorporated into our design. This will make it easier to remove the components of the core that need replacement, such as the graphite reflectors in our design. Figure 3.2.1-2 shows a schematic of how the core components can be removed and reassembled. We believe this feature will make the maintenance of our design rather simple.

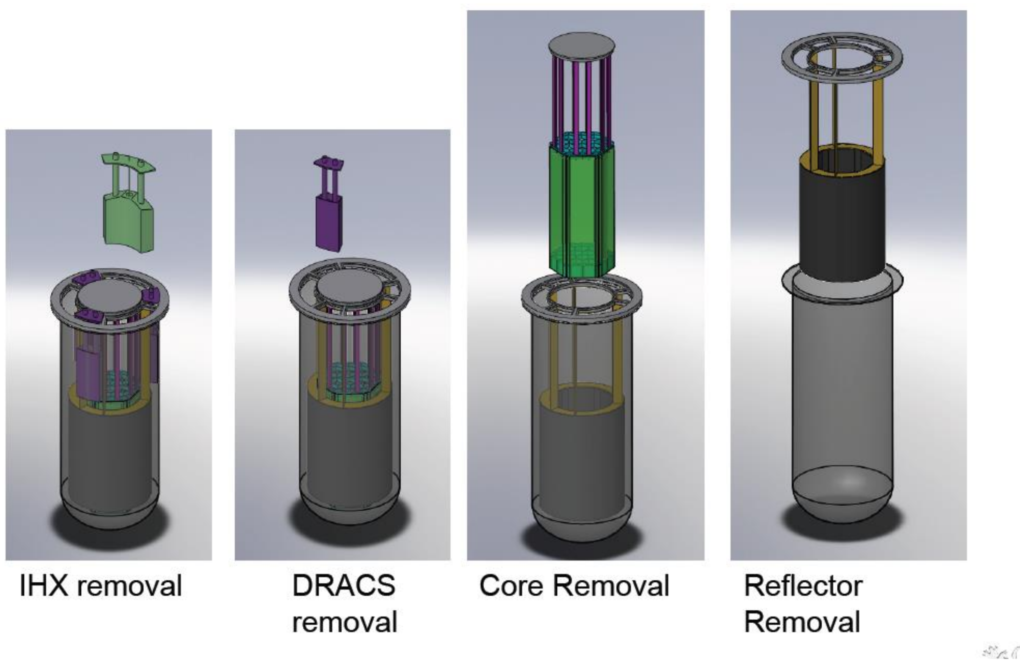

Figure 3.2.1-2: Sm-AHTR Reactor Component removal.

3.2.2 The ThorCon Breeder Design

Incorporating the discussed key features of both the original ThorCon and a Sm-AHTR and our own ideas we were able to design a liquid fueled ThorCon modular reactor. With the design specifications listed in Table 3.2.2-1

ThorCon Breeder Reactor Design	
Fuel Mixture	**UF_4-NaF-BeF_2**
Mol Percentage	**2.5-84.2-13.3**
Breeder Blanket	**ThF_4-NaF-BeF_2**
Mole Percentage	**11.2-76.7-12.1**
Control Rods	**Boron Carbide (B_4C)**
Moderator	**Graphite**
Coolant	**NaF-BeF_2**
Reactor Vessel	**Hastelloy-N**
Electrical Power	**251 MW**
Thermal Power	**643 MW**
Efficiency	**52%**
Core Volume	**117 m^3**
Core Height/Width	**5.96m/5.8m**
Core Fuelsalt Entrance Temp	**800 ^oC**
Core Fuelsalt Exit Temp	**1000 ^oC**
Active Core Volume	**10.55 m^3**
Active Core Height/Width	**3.36m/2m**
Heat Exchanger Type	**Shell and Tube**
Shell side fluid/diameter	**Fuelsalt/ 0.15m**
Tube side fluid/diameter	**Air/0.05m**
Heat Exchanger Area	**21.9 m^2**
Length/Diameter of Tubes	**13.94 m/0.05 m**
Shell Length	**3.92 m**
Shutdown Margin	
Burnup	

Table 3.2.2-1: Important Parameters of the ThorCon Reactor Design (*Subject to change-STC)*

The core of our ThorCon liquid fueled reactor is 5.96 meters tall and 5.8 meters in width. The fuel is in the center of the core of the reactor and is surrounded by a graphite moderator, which is surrounded by an annulus that is made of the thorium breeder blanket. The fuel, moderator, and breeder blanket are all incased in a graphite reflector. Both the fuel and breeder blanket are liquid and is homogeneously mixed with the coolant, NaF-BeF_2. Both the breeder blanket material and the fuel salt are pumped through a chemical processing unit that removes the decay products and releases the fuel salt back into the center of the core from the bottom. As this is happening the fuel salt is being pumped through the heat exchanger and steadily pumped back into the core. Through this process natural circulation occurs and the fuel is continuously mixed so that there are no hotspots due to burn up. In *Section 3.5 Core Analysis* the dimensions and specifications of the reactor core are explained in further detail.

The reactor vessel is made of Hastelloy-N, a nickel-base alloy invented at Oak Ridge National Laboratories (ORNL) as a container material for molten fluoride salts[9]. According to Haynes International, Hastelloy-N is good for continuous operation at temperatures up to 982^{o}C and for intermittent use at temperatures up to 1038^{o}C. Hastelloy-N is also useful for its high resistivity to corrosion and expansion under high temperatures. ORNL tensile testing has shown no signs of embrittlement at a mean temperature of 816^{o}C for long periods of time. Each reactor vessel is placed inside of ThorCon's Silo Membrane Wall, which will help reduce the temperature effect on the reactor vessel [6].

One core produces 642 of thermal power and 250 MW of electrical power. The nuclear plant will have four active cores and one empty reactor vessel so that each core can be easily removed and replaced when its 4 year life span has ended because of the graphite reflectors/moderator needing replacement. This phenomenon is further explained in *Section 3.3 Start Up and Operation Cycle*. The overall plant electrical output is designed to be 1 GWe.

Each reactor has its own set of compressors, heat exchanger (see section 3.6 Thermal Hydraulics), and turbine to produce its electrical output. This plant runs on an open-air Brayton cycle with a bottoming Rankine cycle that combine for an overall efficiency of 52%. The Brayton cycle uses normal room-temperature air at atmospheric pressure as its fluid to keep costs low. This cycle runs on one loop of intercooling, using two compressors, at an efficiency of 83%. Each turbine operates at 89% efficiency and releases the air at a temperature of 370^{o} C, this air is then used to run the bottoming Rankine cycle. This bottoming Rankine cycle adds an extra 79 MW of net power of electrical output and operates at an efficiency of about 24%. The water used to operate the Rankine cycle will also be used in the intercooling system of the compressors to alleviate intercooling expenses.

3.3 – Start Up and Operation Cycle

Though each core will be continuously fueled via breeding of uranium-233 from thorium-232, the graphite reflector will eventually wear out. Other core internals may need to be replaced as well, but less frequently. ThorCon Power estimates that the graphite will need to be replaced every four years. Due to the breeding process and molten salt heat capacity shutting down a reactor of this design will take significantly longer than shutting down a typical light water reactor. Though the control rods can be inserted to remove reactivity, the decay of protactinium will continue to produce more uranium. This will lead to an increase in reactivity again and potentially high levels of uranium. Because of the long shutdown and startup times, we will utilize a five core design. Four cores will run at all times, while the fifth core is undergoes maintenance. When a core reaches four years of operation it will be drained into the now fully refurbished core, and the old core will then undergo maintenance. Thus, a core will be swapped out each year. This process will provide more than enough time to conduct all necessary checks and graphite replacement.

To start up a ThorCon Breeder reactor the core will be previously doped with U-235 at a slightly higher concentration but the same enrichment as U-233 will be under normal operations. (*See 3.4.2 Fuel and Breeder Chemical Composition*). This will provide a smooth startup of the reactor and will provide approximately six months for the thorium breeder blanket to breed U-233 and feed it into the core as the U-235 is completely burned up. After U-235 is burned up the reactor should have already attained an equilibrium between breeding and power output to maintain steady operation.

3.4 – Fuel Specifications

3.4.1 Fuel Cycle

The thorium fuel cycle is to be used for the generation of fissile uranium-233. The complete cycle and the intermediate processes are depicted in figure 3.4.1.

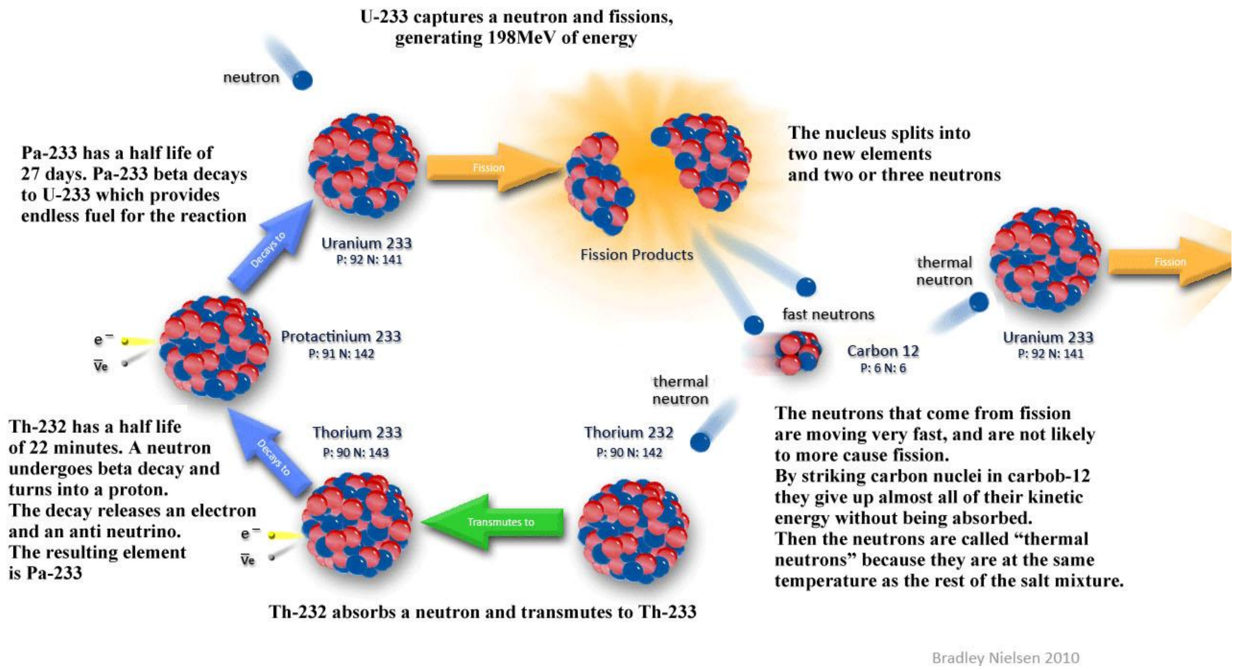

Figure 3.4.1-1: Description of the Thorium Fuel Cycle (http://www.2112design.com/blog/lftr/)

Since the isotope (Pa-233) preceding U-233 has a half-life of 27 days, it will take considerable time for the reactor to reach isotopic, and thereby fuel cycle, equilibrium. Equilibrium is represented by equation (#)

$$U^{233} Fission\ Rate = \frac{dU^{233}}{dt} = \frac{dPa^{233}}{dt} = \frac{dTh^{233}}{dt} = Th^{232}\ Capture\ Rate \quad (1)$$

Utilizing equation (1), the approximate number of density for each intermediate isotope in this breeding cycle necessary for isotopic equilibrium are calculated as

$$N_{U^{233}} \sigma_f \varphi = -\lambda_{Pa^{233}} N_{Pa^{233}} = -\lambda_{Th^{233}} N_{Th^{233}} = N_{Th^{232}} \sigma_c \varphi \quad (2)$$

The results of this analysis are contained in Table 3.4.1

Table 3.4.1 – Intermediate Isotope Inventory

Isotope	Number of Atoms	Atomic Mass	Mass (kg)
Th-233	**3.780e16**	**233**	**1.4625e-8**
Pa-233	**6.758e19**	**233**	**2.6147e-5**

3.4.2 Fuel and Breeder Chemical Composition

In order to create a ThorCon breeder reactor a thorium to uranium to coolant salt ratio needs to be justified. Our team decided not to deviate to far from ThorCon's original fuel salt specifications. With the use of their salt chemical composition, we were able to derive a composition specifically for the fuel mixed with the coolant and another composition for the thorium (breeder) mixed the coolant. Doing this we were able to maintain the uranium to thorium ratio, the uranium to coolant ratio, and the thorium to coolant ratio.

ThorCon expressed through email correspondence (see APPENDIX N) that their fuelsalt composition was based on a few factors, that we designed our fuel and breeder composition to fit. Firstly, their reactor normally use 19.5% enriched uranium, which is U-235 to start the reactor and the same enrichment of U-233 throughout the reactor life. The initial salt composition of an original ThorCon reactor with a volume of 12 m^3 used to generate 557MWth is listed in Table 3.4.2-1. The ThorCon Breeder system we designed uses the same startup enrichment of U-235 at a slightly higher concentration of 2.6% of UF_4.

Table 3.4.2-1 Salt Initial Composition	
Isotope	**Mass (kg)**
Be-9	654.83
F-19	17163.22
Na-23	10579.41
Th-232	14050.10
U-235	654.68
U-238	2652.20

The salt composition is also based on the melting of the heavy metal in the coolant salt, beryllium. According to Lars, of ThorCon Power, the heavy metal fraction needs to stay near 12% to minimize the melting point of NaBe. Doing so in our design allows us to operate at a higher temperature, given that the reactor vessel can withstand it, without boiling the fuelsalt.

ThorCon's normal operation fuelsalt composition is shown in table 3.4.2-2

Table 3.4.2-2 Fuelsalt Mol Percentage	
Isotope	**Mol Percentage (%)**
NaF	76
BeF_2	12
ThF_4	9.8
UF_4	2.2
Total (UF_4-ThF_4-NaF-BeF_2)	100

The first step to creating our fuelsalt and breeder blanket compositions was determining the UF_4 to Th_4 ratio that ThorCon operates on which is simply dividing their mol percentages which gives 0.22. This was done to have a standard to measure our fuel and breeder components chemical composition by. From there we separated ThorCon's mol percentage values for just the fuel salt, UF_4-NaF-BeF_2, and the breeder blanket, ThF_4-NaF-BeF_2. With the numbers separated for each component they were converted to give a total 100% for each mixture. This was done simply by using the ratio of the fuel to the coolant mol percentage and converting it to equal 100%, the same was done for the breeder blanket. An example using the fuel salt numbers are shown in equations (1) and (2).

First, calculate the required percentage of UF_4 needed.

$$\frac{UF_4\%}{NaF - BeF_2\%} = \frac{x_{UF_4}}{100\%} \tag{1}$$

$$\frac{2.2\%}{76\% + 12\%} = \frac{2.2}{88} = \frac{x_{UF_4}}{100}; x_{UF_4} = 2.5\%$$

Second, the amount of coolant (NaF-BeF_2) required in the fuelsalt mixture is calculated. N

$$x_{NaFBeF_2} = 100 - x_{UF_4} \tag{2}$$

$$x_{NaFBeF_2} = 100 - 2.5 = 97.5\%$$

Third, calculate the amount of each component of the coolant is needed using ThorCon's original amounts.

$$\frac{NaF\%}{NaFBeF_2\%} = \frac{x_{NaF}}{x_{NaFBeF_2}} = \frac{x_{NaF}}{97.5\%} \tag{3}$$

$$\frac{76\%}{88\%} = \frac{x_{NaF}}{97.5\%}; x_{NaF} = 84.2\%$$

$$\frac{BeF_2\%}{NaFBeF_2\%} = \frac{x_{BeF_2}}{x_{NaFBeF_2}} = \frac{x_{BeF_2}}{97.5\%} \tag{4}$$

$$\frac{12\%}{88\%} = \frac{x_{BeF_2}}{97.5\%}; x_{BeF_2} = 13.3\%$$

With these equations the mol percentages we needed to create our breeding system was calculated also the percentage of each isotope was found for MCNP purposes. Table 3.4.2-3 shows our fuelsalt and breeder blanket compositions and their overall isotopic mol percentages that were derived.

Table 3.4.2-3 Fuelsalt and Breeder Blanket Mol Percentages			
Fuelsalt	Mol Percentage (%)	**Breeder Blanket**	Mol Percentage (%)
UF_4 (U-235) UF_4 (U-233)	2.6; U=0.8 F=1.8 2.5; U=0.5 F=2.0	**ThF_4**	11.1; Th=2.2 F=8.9
NaF	84.2; Na=42.1 F=42.1	**NaF**	76.8; Na=38.4 F=38.4
BeF_2	13.3; Be=4.4 F=8.9	**BeF_2**	12.1; Be=4.0 F=8.1

Our chemical composition met the requirement of the ThorCon fuelsalt and its UF_4 to ThF_4 ratio. Our heavy metal percentages remained near 12% and our fuel is capable of operating at 19.5% to 20% enrichment. Our ratio of fuel to breeder remained at 0.22. With this we are able to proceed with neutronics and modeling our system in MCNP6.

3.4.3 Chemical Separation

The fuel used in this reactor is Uranium-233 which as previously discussed must be bred by the absorption of a neutron by Thorium-232 which will lead to a decay. In our reactor the thorium salt is in a separate “breeding blanket” from the fuel which will be located in the central core region. When Uranium-233 is bred it must then be separated from the thorium and protactinium found in the breeding blanket. We can use a process developed by oak ridge national laboratory[12].

When ThF_4, PaF_4, and UF_4 are all exposed to Florine only the uranium will readily bond with it. Thorium-fluoride and protactinium-fluoride only allow for a maximum of four fluorine atoms to bond to them, whereas uranium will readily accept two more to form uranium-hexafluoride, a gaseous form of uranium used in enrichment processes. By running Florine through the fuel salt we can readily separate out the Uranium by simply venting off the uranium gas for storage and reprocessing. Hydrofluoric acid can be used to strip the extra Florine to convert the UF_6 to UF_4. Beyond this step a lithium bismuth stream could be fed into the breeding salt stream to remove protactinium from the breeder salt. This could be useful to ensure the Pa^{233} doesn’t absorb a neutron and become Pa^{234}.

3.5 – Core Analysis

3.5.1 Neutronics

Neutronics or neutron transport is the study of motions and interactions of neutrons with materials. We implemented Monte Carlo Particle Code (MCNP6) version 6 to analyze the neutron transport of our reactor design. The code deals with transport of neutrons, gamma rays, and coupled transport (transport of secondary gamma rays resulting from neutron interactions). The MCNP code can also treat the transport of electrons, both primary source electrons and secondary electrons created in gamma-ray interactions.

Factors such as the breeding ratio of our reactor, neutron and thermal fluxes across the core, the criticality and burn up, reactivity insertions in the core, along with other parameters of our design have been calculated and analyzed using MCNP6. Safety analysis of our core was also conducted using MCN6.

3.5.2 Reactor Core

The 5.71-m high by 2.75-m radius reactor pot contains molten fuelsalt that is homogeneously mixed with the coolant, and graphite cylindrical block for neutron moderation and reflection. The core calculated size was based on the geometric and material buckling factors shown in equation (5), in which both must be equal for criticality (see APPENDIX A).

$$B_g^2 = \left(\frac{\pi}{H}\right)^2 + \left(\frac{2.405}{R}\right)^2 = B_m^2 = \frac{\nu * \Sigma_f - \Sigma_a}{D} \qquad (5)$$

In this equation, the critical radius, *R*, and the critical height, *H*, affect the geometric buckling and the k_{eff} of the critical core. The hand calculation of the geometric and material buckling yield a critical radius was 2.75 m. However, in MCNP6 a radius of 2.9m was used to reach criticality which is not far off the radius calculated from the hand derived radius. It was determined that the small discrepancy in radiuses was small enough to proceed with the use of MCNP for criticality calculations.

The second step in utilizing MCNP for core design was to find k_{inf} for an ideal homogenized mixture of fuel salt, graphite, and thorium. To do this, we created unit cells with borders as an input into MCNP. The full code for this can be found in the appendix (see APPENDIX E), and a diagram is shown in Figure 3.5.2-1. The hastelloy-N was not modeled, so that the neutron transport could be calculated as bare system in this scenario. The k_{inf} that results from this configuration is 1.1.

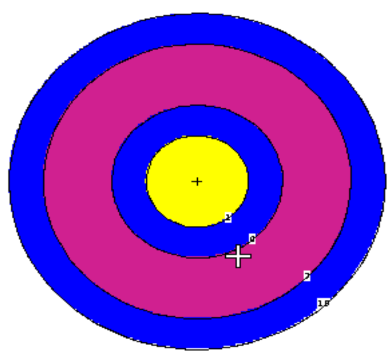

Figure 3.5.2-1: Fuel salt (yellow), graphite (blue), and thorium blanket (burgundy).

Since kinf is adequately high enough to keep the reactor powered, the next step was to plan the layout of the core. To maintain this criticality our design must breed enough uranium 233 (see section *3.4 Fuel Specifcations)* to maintain the required fission rate for our power output. To achieve an equilibrium of fuel burn up in our core we are assuming that the overall flow of the fuel salt is turbulent. This assumption is based on the high operating temperatures (which leads do a less dense fuel salt), the heat sink of the reactor, and pump flow of the fuelsalt in and out of the core (see section *3.6 Thermal Hydraulics).* Turbulent flow decreases the probability of hot spots occurring in our reactor core, with this, the mixing of the fuel salt improves the efficiency of heat and mass transfer throughout the fuel salt region.

Since the design of our system is a breeding design, our k_{inf} is designed to be near critical (k=1) to limit excess reactivity in our core. We are limiting the excess reactivity in our system because of the constant pumping of fuel salt in and out of our active core region and we want to minimize the chances of critical salt flow outside of the core. In a non-breeding reactors, referring to section 1 in *University of Wisconsin's Critical Experiment manual*, the reactor operates with a k_{inf} average around 1.3 to have a feasible amount of excess reactivity. However, our breeding system is designed to operate with a k_{inf} of 1.01. This is done so the fission poisoning in the fuelsalt is projected to make the fuel salt subcritical with a k_{eff} of 0.98.

We then analyzed the breeding ratio of our reactor design. The breeding ratio is the number of reactions in the fertile material (Th-232) to number of fissions occurring in the fissile material (U-233)17. The breeding ratio in most breeding reactors aim for 1 according to the *Roadmap Design for Breeding Ratio section 2*. Our system can achieve a breeding ratio of 1.06 reactions/fissions, which we calculated using MCNP6. To achieve this breeding ratio our core design geometry needed modification. With too much graphite, the breeding ratio is extremely. This is because there are more fissions occurring per neutron than breeding reactions occurring per neutron in the thorium blanket. If there were too little graphite, the fissions per neutron would decrease significantly thus causing the criticality to decrease. This leads to the breeding ratio improvement in our system relying on the location of the thorium blanket in respect to its nearness of the fuelsalt core. At a distance of 1 meter away from the fuel salt, the breeding ratio suffered. However, at a distance of .25 meters, we achieved a breeding ratio of 1.04 at our bulk operating temperature 875^{o}C. Once the adjustment of the breeding blanket was done the effect of temperature on the breeding ratio of our system was analyzed. This is shown in figure 3.5.2-2.

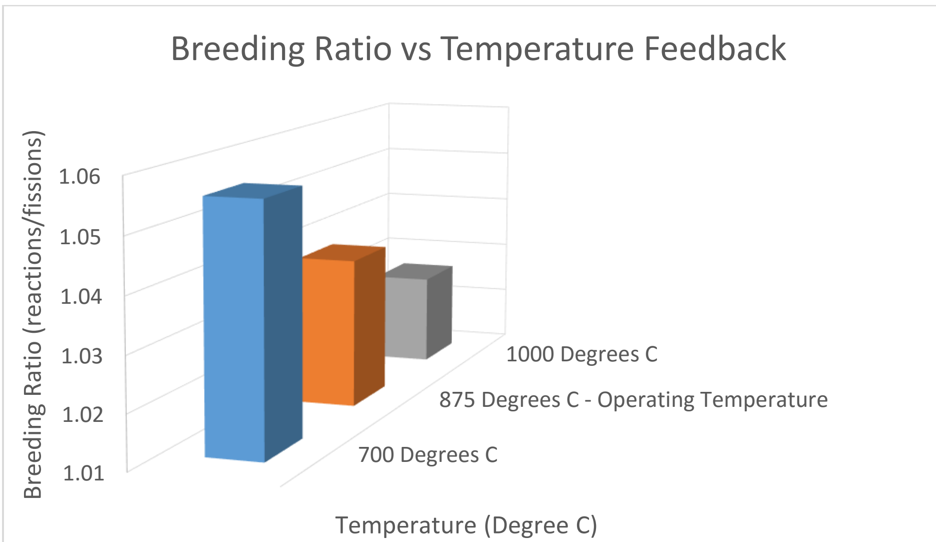

Figure 3.5.2-2: Breeding ratio of the ThorCon Breeder

Figure 3.5.2-2 shows that as the temperature of the overall core increases the breeding ratio decreases. At a bulk temperature of 1000^{o}C the breeding ratio drops to less than 1.03. The decline in the breeding ratio is expected and wanted. In overheating scenarios, the breeding ratio will continue to drop and thus fall below 1 and lead to a subcritical state where the amount of U-233 produced is insufficient in keeping the core critical.

With the core design finalized, the next step was to determine the core's shutdown margin. To do so we first found the required negative reactivity addition needed to drop the k_{eff} of the system. To drop the k_{eff} of our system we will use the insertion of control rods at different bank heights to achieve our goal. With the core's limited reactivity, it should be easy to achieve a shutdown margin when the reactor is in a subcritical state. Table 3.5.2-1 shows the shutdown margin found from MCNP6 when the control rods are added to the system and when the control rods are fully inserted.

Table 3.5.2-1: Shut Down Margin of the ThorCon Breeder	
Shutdown Margin	**K**
K_{inf}	1.08
Addition of Rods	
K_{eff}	0.98
Negative Reactivity	1.2%
Full insertion of rods	
K_{eff}	0.92
Negative Reactivity	8%
Shutdown-margin	8%

Table 3.5.2-1 shows that once the rods were added the k_{eff} of the core was reduced to 0.98 and with full insertion, k_{eff} drops to 0.92, which is a reasonable k_{eff} for normal shutdown operations.

Table 3.5.2-2 shows the multiplication factor, k, of the reactor at various bank heights and at normal operation temperatures (875°C) and at 700°C, which is considered cold start up temperature for our reactor type.

Table 3.5.5-2: Neutron Multiplication Factor at several control bank heights in the graphite.

Measured from top to the bottom of graphite	K (at 700°C)	K (at 875°C)
100 cm	1.07	1.04
150cm	1.06	1.03
200 cm	1.05	1.01
285cm	.97	0.92

Figure 3.5.2-3 shows the final geometry of the core with the control rods added. The initial design was to have four control rods in the center of the fuelsalt core. However, we were advised that keeping the control rods stable in the fuelsalt turbulent flow might be difficult. Reconsidering the placement of the control rods led to moving them to the graphite moderator. With this new control rod design, holes will need to be inserted in to the graphite to hold the rods. There are a total of 28 rods in the inner graphite moderator. The number of rods increased because they are outside of the active core region and are not as effective as they were when placed in the center of the fuelsalt.

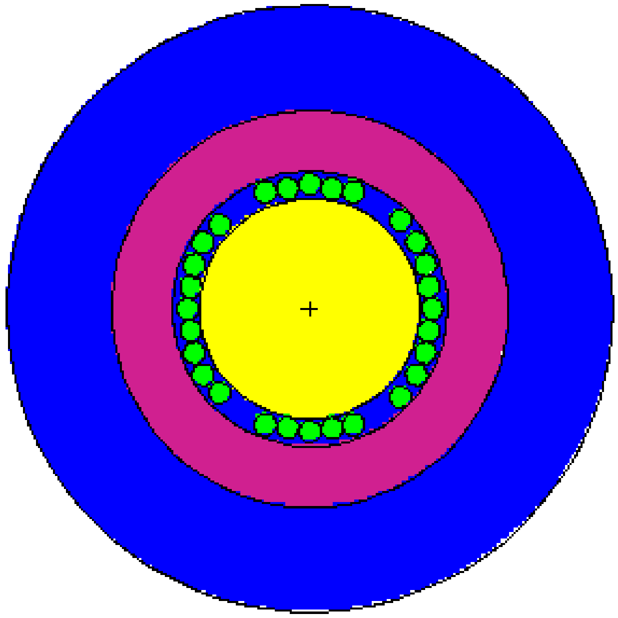

Figure 3.5.2-3: ThorCon Breeder with control rods (green) inserted in the graphite (blue

3.5.3 Fuel to Graphite ratio

One of the most important design components of a MSR core is the fuel to graphite ratio. The fuelsalt to graphite ratio was calculated in two ways in this section for our ThorCon reactor design. One way was defining the volume ratio of the fuelsalt to graphite in the entire core. This was calculated to be 0.1 or 10% ratio. The second method was using the thermal utilization factor from the six factor formula. The resonance escaper probability is determined largely by the fuel-moderator ratio arrangement and amount of enriched U-235 (19.5%) [20]. Since the section is focused on the fuel to graphite ratio and fission of the fuel depends on the moderator (graphite), the thermal utilization factor calculation was based on the ratio of the absorption rate of neutrons in the fuel to the neutrons absorbed in the graphite moderator region. Using MCNP6 absorption for both fuel and moderator regions, a thermal utilization factor for the fuel to graphite ratio was calculated to be 0.19 or 19% at 700 C. Figure 3.5.3-1 displays the decrease in k-effective due to increase in fuel to graphite [20]. According to “Nuclear Power Training” large core fueled with low-enriched fuel, there is an optimum point above which increasing the moderator-to-fuel ratio decreases k-effective due to the dominance of the decreasing thermal utilization factor. For our system as the temperature increase the fission rate in the fuel decreases, this leads to an increase in the fuel to moderator ratio (thermal utilization factor). This phenomenon plays in our favor because as the thermal utilization factor increases with temperature, the k_{eff} of the system decreases which serves as a passive safety mechanism in overheating scenarios.

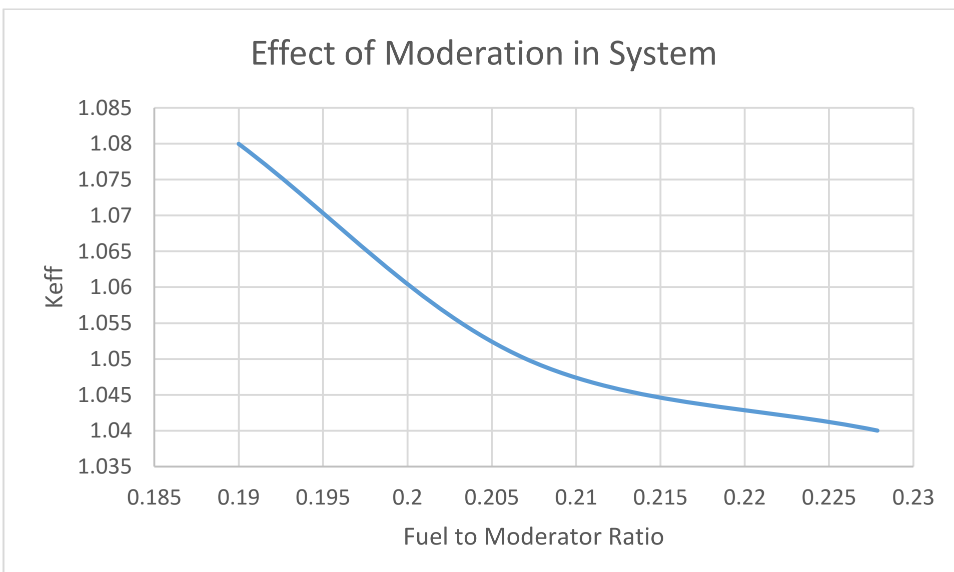

Figure 3.5.3-1: Fuel to Moderator ratio effects on the K_{eff} of the system.

3.5.4 Reactor Core Analysis

Using the methodology of MSR core calculations, this section will carry out results for the multiplication factor, neutron flux distribution, and power distribution. All of the calculations are based on steady state conditions. The neutron flux distribution for the reactor system is analyzed radially and axially for the entire core. To calculate the multiplication factor the percentages of the material compositions, the enrichment of U-233 and the geometry buckling are taken into consideration. The power distribution is found by dividing the core into 4 regions with 5 different power levels. Figure 3.5.4-1 shows the entire core with the addition of the Hastelloy-N reactor vessel.

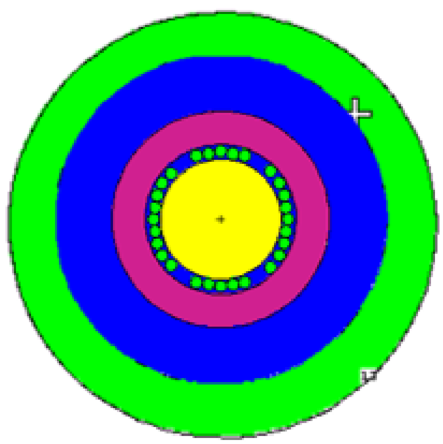

Figure 3.5.4-1: Displays the entire reactor core.

3.5.4.1 Effects on the Neutron Multiplication Factor

This section shows the results of a comparison between the burnup of the fuel and k_{eff} of our system at various temperatures. The change in density of the fuelsalt with the increase in temperature is a significant contributing factor to the change in $k_{eff.}$ As temperature increase the density of the fuel salt decreases with expansion of the fuelsalt. As the density decreases in the fuel, less collisions are likely to occur, resulting in a decrease in k_{eff} as seen in figure 3.5.5.1-1. According to Duderstadt in *Nuclear Analysis*, an increase in temperature reduces densities, macroscopic scattering, and absorption cross sections [8].

The burnup of the fuel within our system is dependent upon the enrichment of the fuel salt. The fuelsalt is 2.2% uranium, which is comprised of U-233 and U-238. Uranium-233 is enriched to 19.5% based on the original ThorCon reactor design. The burnup results of the ThorCon reactor was found using MCNP6.

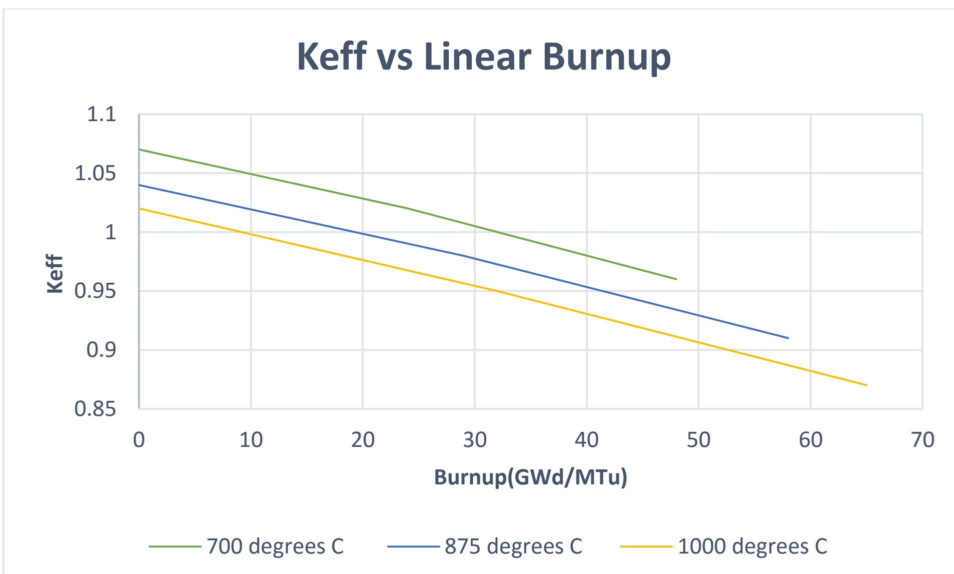

Figure 3.5.4.1-1: Displays the k-eff versus burnup time in the ThorCon Breeder.

K_{eff} was calculated with controls rods inserted in at a position of 285 cm from the top. The position was taken as the average value of the control rod positions for the whole cycle, the k-effective value was found to be 0.99 at operating temperature. When the reactor operation mode is on full power at 642 MW of thermal power, the value of the k_{eff} decreases as in figure 3.5.4.1-2. From figure 3.5.5.1-2 the reactor is subcritical after 60 days at our operating temperature. This means that if our breeding system was unable to breed more U-233 the reactor would only last for approximately 2 months at full power. This 2-month window serves as a safety feature timeframe, if the chemical separation plant fails or the something is amiss in the decay tank used for the thorium decay. Depending on the operation and maintenance projected time, if more than 2 months is needed the new operation temperature for our system would be approximately 700 degrees C. At 700 degrees C, we could attain an additional 20 days.

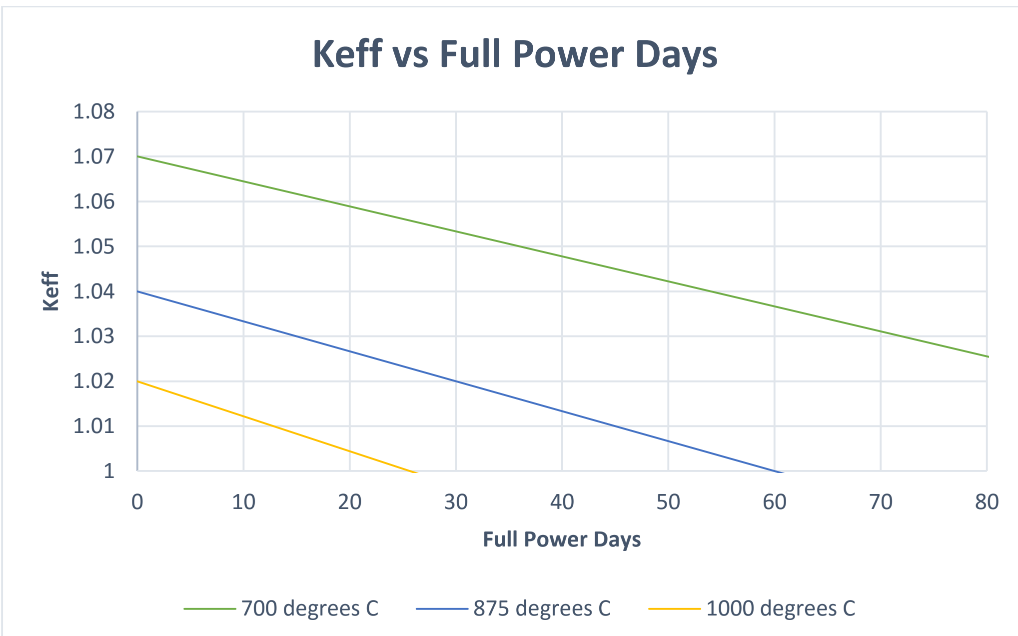

Figure 3.5.4.1-2: Displays the k-eff at different operating temeperatures.

One ultimate challenge faced in the design of the ThorCon Breeder system is starting up the reactor core. ThorCon Power reactor designs are based upon start up with fresh LWR sources; however, their design is arranged with channels and hexagonal solid fuel assembly. Our breeding assembly could use the same approach with fresh LWR sources, but further research will have to be conducted for exact calculations for the number of sources and the behavior of such sources in a homogenized breeding system such as ours. However, we calculated a startup time for our system to reach an equilibrium between U-233 production and steady full-power operation.

The timeframe for a ThorCon Breeder to reach equilibrium will be based on initially loading the reactor with U-235 enriched at 19.5% at 2.6% of uranium in the fuelsalt composition. The equilibrium time will serve as the time needed for U-235 to burn up to a k_{eff} of 0.98 while U-233 is bred and fed into the system to reach the breeding ratio needed to operate steadily at full power. Using MCNP6 burn card technology, the time of equilibrium will approximately take 6 months. Figure 3.5.4.1-3 displays the reactor equilibrium time versus the k_{eff}.

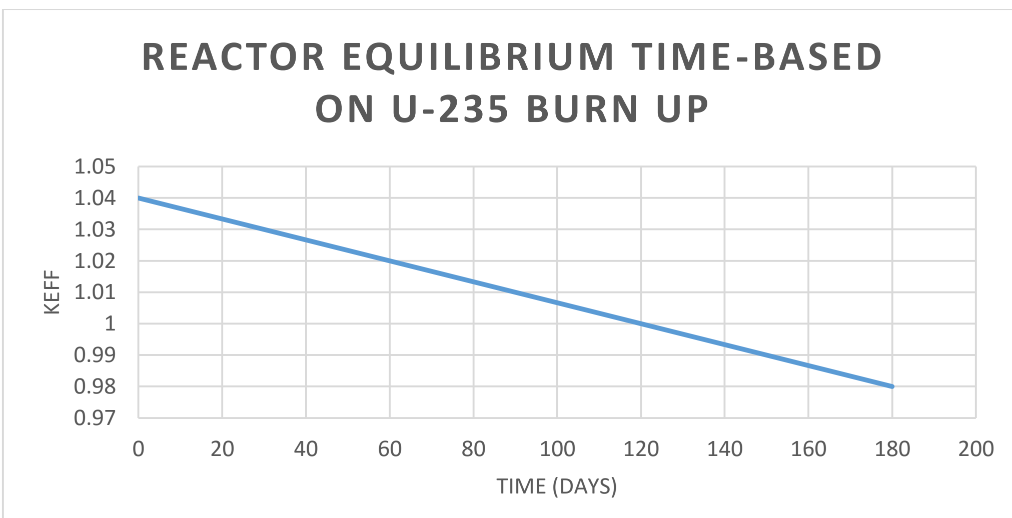

Figure 3.5.4.1-3: Displays the time U-235 allows for the system to reach an equilibrium.

3.5.4.2 Neutron Flux Distribution

The neutron flux distribution of the ThorCon Breeder is dependent upon the heat generation in the reactor fuel. The neutron flux is primarily a function of fission-reaction rate at a specific point in the fuel or core [9]. The fission-reaction rate at steady state conditions is based upon our breeding conversion. The distribution is also very dependent upon the energy generation in the core and is of major importance in the core design. Our design is a cylindrical core design in which a flux distribution in a cylindrical core is given by equation (6).

$$\varphi = \varphi_{CO} * cos\frac{\pi z}{H_e} * J_0\left(\frac{2.4048r}{R_e}\right) \tag{6}$$

The flux, ϕ, in equation (6) is a cosine function that is dependent on the position in the core axially, H_e, and radially, R_e. In figure 3.5.5.2-1 the flux is displayed in the axial direction as a cosine curve. The flux distribution is without control rods in assuming that the reactor is operating in steady state at full power. The axial flux distribution spans from inner surface of the bottom of hastelloy-N vessel to the top of the fuel salt, which is essentially from top to bottom of the active core neglecting the reactor vessel. Here the peak can be found near the center of the fuelsalt core.

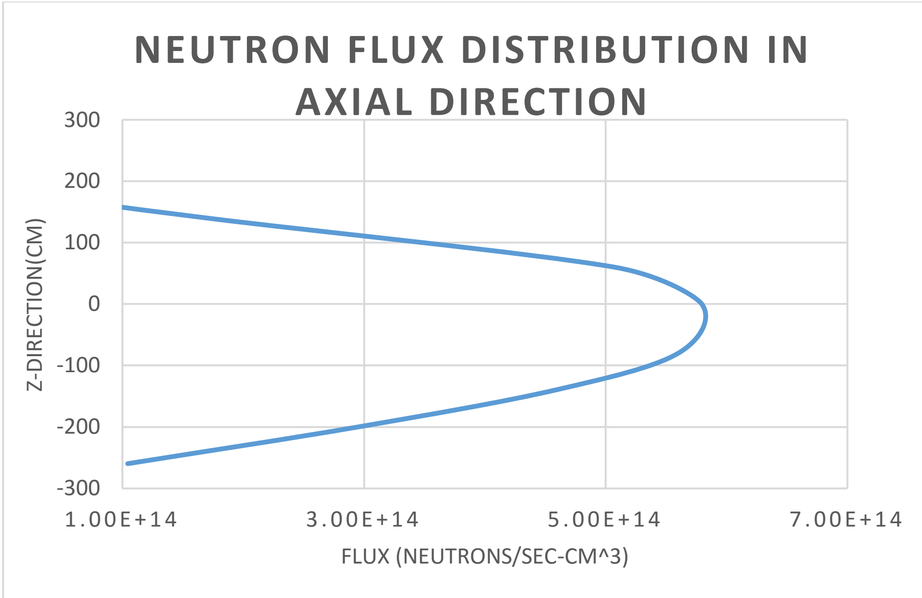

Figure 3.5.4.2-1: Displays the axial neutron distribution of the ThorCon Breeder

Examining the neutron flux in the radial direction, the flux ranges from center of the fuelsalt core to the inner radius of the hastelloy-N vessel region. As in the axial direction the flux in the radial direction is also a function of cosine. The effective radius of the core is approximately 190 cm. However, the effective radius of the fuel is 100 cm. Figure 3.5.4.2-2 shows a neutron flux decreasing continuously with radial distance from the center.

A normal cosine distribution appears to be a smooth curve, known as flux flattening [16]. However, in figure 3.5.4.2-1 our cosine curve is not a smooth curve. Our theory is that flux distribution is non-uniform because of the location of our fuel. Typically, the fuel assembly is located throughout the core, in solid fuel assembly that is. In the ThorCon Breeder the fuel location is in the center of our reactor surrounded by the other elements in our reactor core. If the fuel was distributed throughout the core as in plate or tube designs, the fission-reactions will be distributed more across the reactor thus, the neutron flux will also be distributed uniformly and lead to flux flattening [16]. Hence, power of a reactor can be increased without overheating of fuel in any location. This referred to as flux flattening. The technique used to help flatten the flux was the addition of reflector graphite outside the region of the thorium blanket. The use of reflectors brings the about a substantial decrease in maximum flux to initial flux ratio, facilitating flattening of the flux [16]. However, to achieve a smooth cosine curve, using the flux flattening method of control rod insertion in the fuel area would be needed. But our design control rod insertion is in the graphite region, hence as seen in figure 3.5.4-1 the location of the fuel rods inserted in the graphite moderator region is located outside of the fuel salt region of 100 cm in which the curve appears to flatten out. In conclusion flux flatten technique works, but not for our fuel salt region because of the location of control rods. Other techniques used for flux flattening are variations of fuel enrichment, fuel-to graphite ratio, neutron poisons, and adjusting the radius and effective height of the core. These techniques were not implemented in our design due to the effect on other parameters, such as the breeding ratio, geometrical buckling, etc.

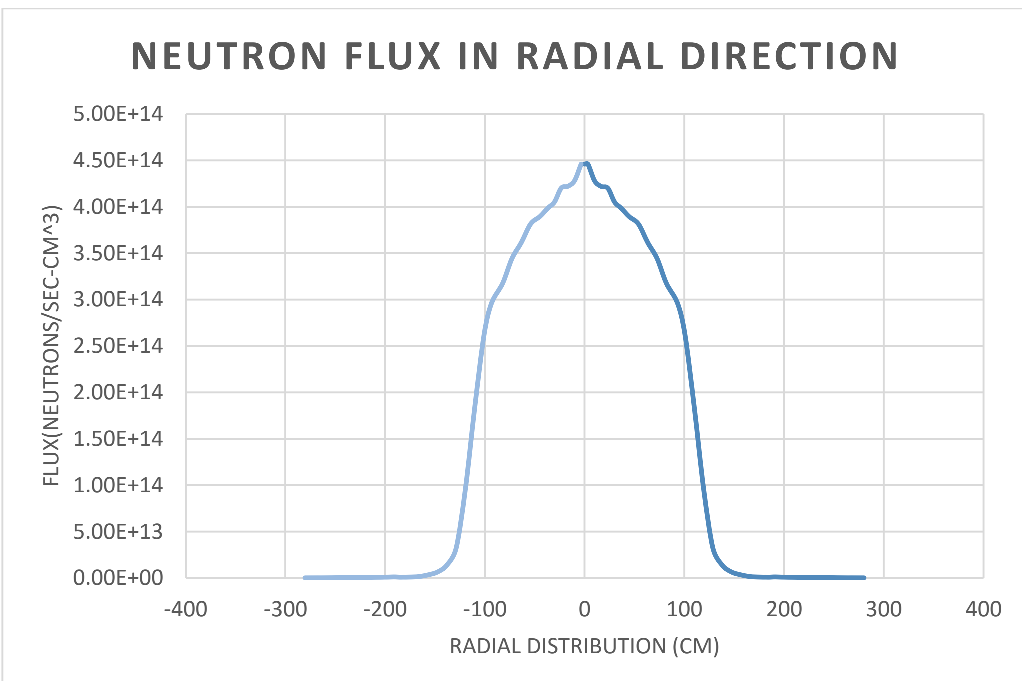

Figure 3.5.4.2-2: Displays the neutron flux in the radial direction of the ThorCon Breeder

3.5.4.3 Power Distribution

This section will cover the power distribution of our system. The power distribution is definied by the conversion of the neutron flux distribution of the core. According to *Relationship Between Neutron Flux and Reactor Power,* in an operating reactor the volume of the reactor is constant. If the reactor volume and macroscopic cross section are constant, then the neutron flux and reactor power are directly proportional. In 3.5.4.3-1, using the MCNP6 cross section plotter, the total neutron cross section for the fuel region was plotted for our system. The neutron flux for a given power level will increase over a period of months due to burnup of the fuel and resulting decrease in atom density and macroscopic cross section. Figure 3.5.4.3-2 displays the increase of the neutron flux proportionally to burn up in our system.

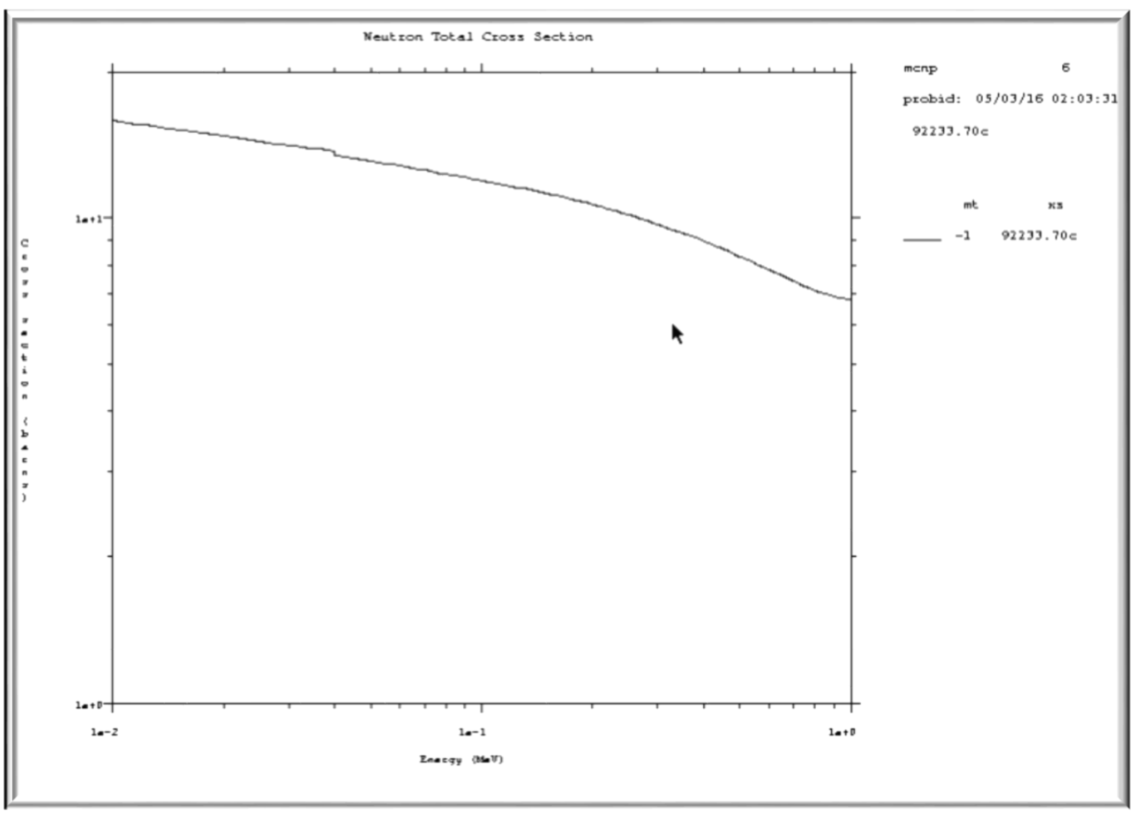

Figure 3.5.4.3-1: Displays the constant neutron cross section plotter for the fuel region in terms of Energy (MeV).

In figure 3.5.4.3-1 the abscissa is labeled Energy (MeV), the ordinate is labeled as the "Neutron Total Cross Section" in a unit of barns. The plot displays a constant decreasing neutron cross section in the fuel region as described previously.

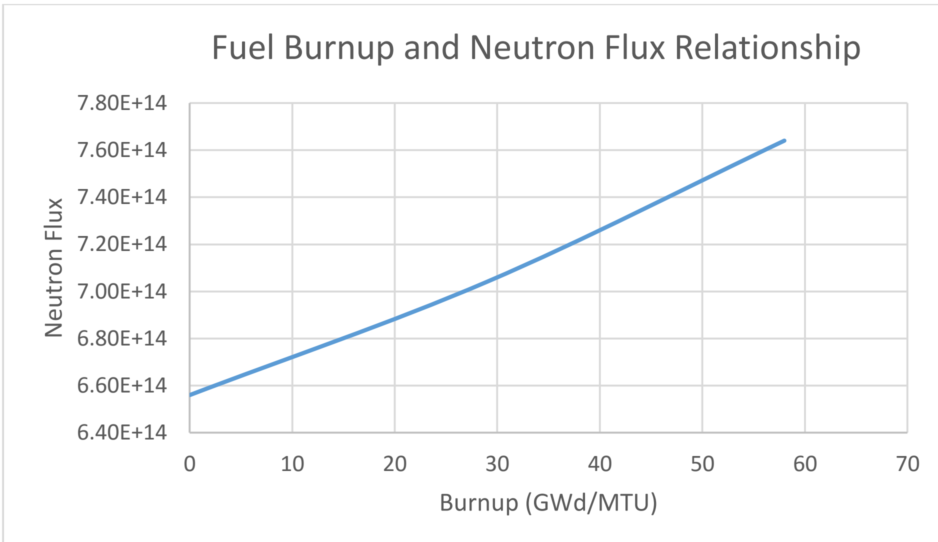

Figure 3.5.4.3-2: Displays the relationship of fuel burnup and neutron flux[20]

The power distribution of our entire system should equal to approximately 642 Megawatt of thermal power. Figure 3.5.4.3-3 is the power distribution of our system in the axial direction in reference to 1 cm radially, from the top of the core to the bottom of the thorium blanket region. To calculate the entire power distribution across the core, we used the trapezoid rule [15]. Essentially the method is integral worth method. The integral method gives you the total power under the curve in a particular region of interest. See APPENDIX C for complete calculations.

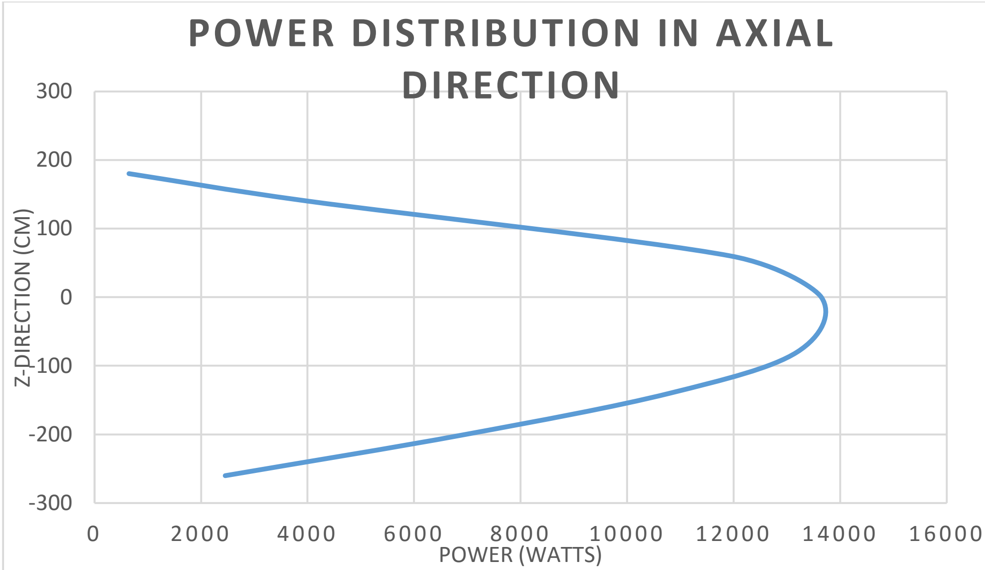

Figure 3.545.3-3: Displays the power distribution in the axial direction.

3.6 – Thermal Hydraulics

After the size, geometry, and power output of the ThorCon Breeder was finalized, work on removing the heat from the system could begin. To stay near ThorCon Power's original designs, it was necessary to set certain parameters that we would maintain to arrive at an electrical power output that is near the original ThorCon reactor. Due to the intricacy of a breeding reactor, a fluid flow schematic was developed to help in designing the primary loop required to remove heat from the system. In this, the mass flow rate of the fuel was found based upon the heat transfer required from the primary loop. Our thermal hydraulic design also features a bottoming cycle that creates an additional output from the waste of the primary loop. This creates a cheap addition of energy for the plant. Based on the neutronics, the core parameters were the driving force in the design of the heat exchanger used in our system.

To stay near to the power output of the original ThorCon Power reactor, an electrical output of 250 Megawatts was set for the ThorCon Breeder. In order to meet this demand, it was decided that a Brayton-cycle would be efficient. In choosing this power cycle, the cooling gas was chosen based on cost. Air was chosen as the cooling gas being that is easily obtainable and easy to cool and work with. Also, choosing air as our cooling gas meant that the Brayton cycle could be open, being that it was essentially clean air it could be let back into the atmosphere once it went through the cycle, this also alleviates some of the cost. However, instead of dumping the waste air, it was decided that the air removed from the system could be used to power a bottoming cycle. This bottoming cycle is a Rankine cycle that would use the heated air to produce an additional amount of electricity. Once the Brayton-air cycle and bottoming Rankine cycle was set, various types of heat exchangers were discussed. The decision was made to use a shell and tube heat exchanger, this was based on the research conducted in molten salt reactors. The SM-AHTR design, discussed in *Section 3.1 ThorCon Power Literature Review* uses a shell and tube heat exchanger that sits on top of the active core. The SM-AHTR also implements natural circulation to circumvent pump work. Parts of this design will also be implemented into the ThorCon Breeder thermal hydraulic system. Figure

3.6-1 shows the fluid flow schematic of the ThorCon Breeder once all thermal hydraulic components were finalized.

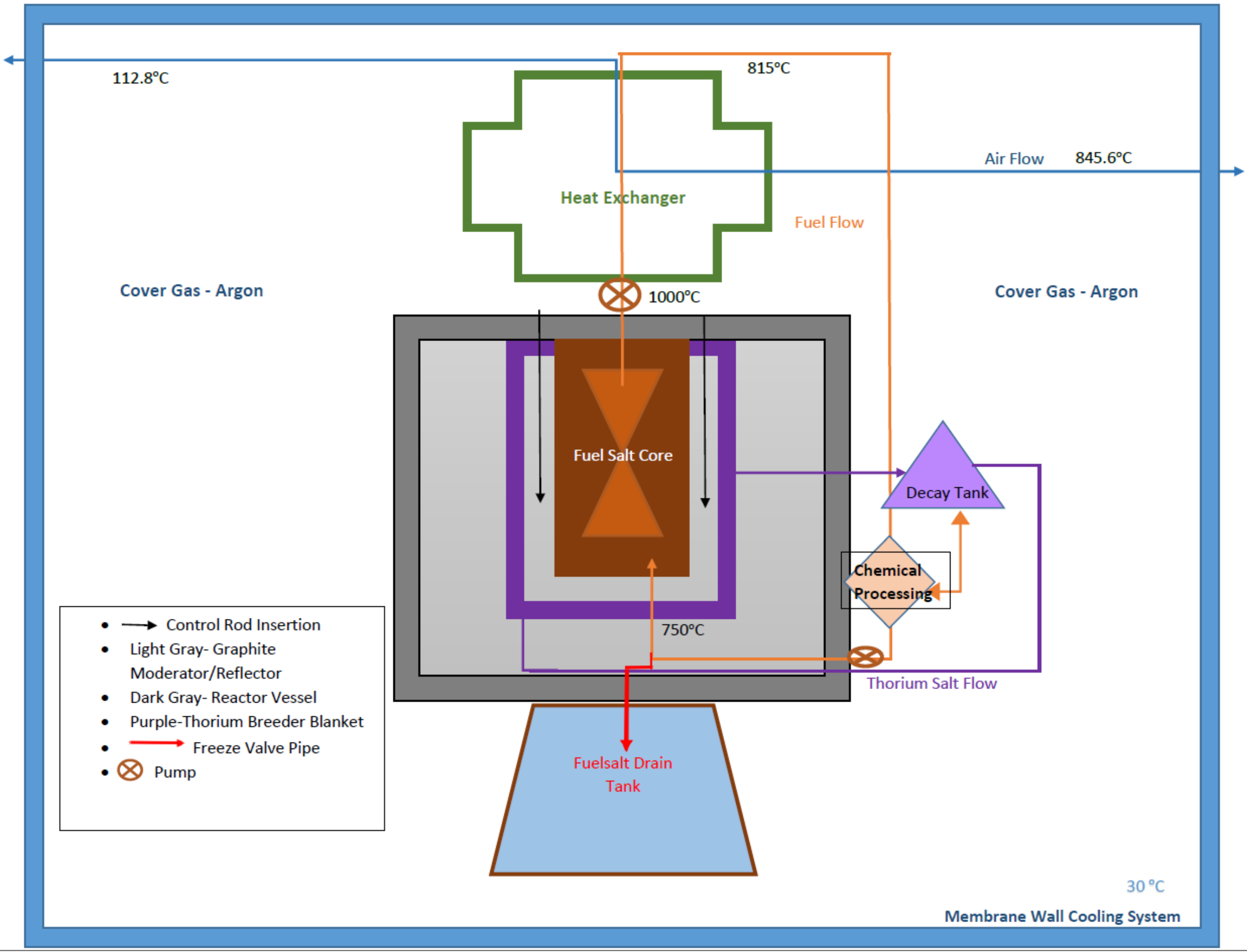

Figure 3.6-1: Displays the fluid flow of a single ThorCon Breeder System

3.6.1 Primary Loop Brayton Cycle

From the research conducted on the SM-AHTR designs we realized that the removal of heat from the reactor depended on a Brayton cycle. In the SM-AHTR design there is typically a 2 to 4 loops of heat removal, which is usually comprised of a fuel to coolant salt step and then the coolant salt to the cooling gas in the heat exchanger. For the ThorCon Breeder however, only one loop is used, fuelsalt to air. Being that our coolant is homogenously mixed with our fuel the coolant salt is readily heated and can easily transfer heat to the air in the Brayton cycle. Also, once the fuelsalt leaves the core, it also leaves the moderator (graphite) which renders it subcritical. As the salt flows through the heat exchanger its criticality will readily drop and only rely on short-lived fission products decay. This alleviates some of the danger of radiating the air that flows pass the hot fuelsalt in the heat exchanger.

With a single transfer of heat from fuel to air, we needed to make sure that we were able to maximize the amount of power that could be obtained from the open-air Brayton cycle. In order to do so, it was decided that the air would be precooled in room to a temperature of 20^{o}C. From there the air would undergo a stage of intercooling, this would essentially re-cool the air at a high pressure before it enters the heat exchanger.

In order to begin calculations on the open-air Brayton cycle we set design parameters based on research conducted on various types of air brayton cycles. As a starting point we modeled a basic brayton cycle with a primary heat exchanger, turbine, and a stage of intercooling. In order to get an idea of the work needed to creat 250 MW of electricity, we used examples from textbooks, such as Nuclear Heat Transport by El-Wakil. From this we determined a mass flow rate of air need, and the initial temperature of the air, and the temperature of the air entering the turbine.

Being that we wanted to stay true to the SM-AHTR and the original ThorCon design, we designed the Brayton cycle to operate on atmospheric pressure air (101.3 kPa) which was feasible based off researching Brayton cycles. The initial mass flow rate of air was based on the power output needed, 250 MW. Essentially we converted this electrical output to a rough estimate of thermal power which was approximately, 675 MW-thermal. From this the mass flow rate was calcualated using equation (7).

$$Q = \dot{m}c_p(T_5 - T_1) \tag{7}$$

Where Q is the thermal power, cp is the specific heat of air, and T_5 and T_1 are the outlet and inlet temperatures of the open air brayton cycle, respectively. The temperatures of the outlet and inlet were chosen to be 800°C and 20°C to maximize the heat sink in the system and to bring the temperature of the air as close as possible to the temperature of the hot fuelsalt, which is 1000°C. From these estimates the initial estimates the mass flow rate was initially 1035 kg/s. With the addition of the efficiency of the turbine and compressors, and iteration of the Brayton cycle, the final mass flow rate of air was found to be 865 kg/s. This number seemed feasible base on the amount of power we are trying to produce from a single heat transfer loop.

The efficiency of the compressors and turbine were also based on examples derived from textbooks and then were altered through out iterations to get the best possible thermal efficiency of the open-air Brayton cycle. The final efficiency of the compressors is 83% and the turbine is 85%, from research and class literature these numbers seem ideal for compressor and turbine sizing.

With the final iterations of the open-air Brayton cycle the mass flow rate remained at 865 k/s and the efficiency increased to help increase the overall efficiency of the cylec. The pressure ratio of the system was also calculated based on the inlet and outlet temperatures of the cycle, which was calculated to be 9.695 this was a significant factor in our design. In the beginning the assumption of the pressure ratio as 6.5 was not feasible in uping the thermal efficiency and the overall electrical output. Table 3.7.1-1 shows the final parameters of each state of the open-air Brayton cycle. This information can also be seen in the schematic shown in figure 3.7.1-1.

Table 3.7.1-1 Thermodynamic States of the Open-Air Brayton Cycle

State	Pressure (kPa)	Temperature (^{o}C)	Enthalpy (kJ/kg)	Entropy (kJ/Kg*C)
1	101.3	19.85	293.4	5.678
2	315.5	112.8	387	5.629
3	315.5	19.85	293.4	5.352
4	982.3	112.8	387	5.303
5	982.3	799.9	1130	6.396
6	101.3	370	652.9	6.482

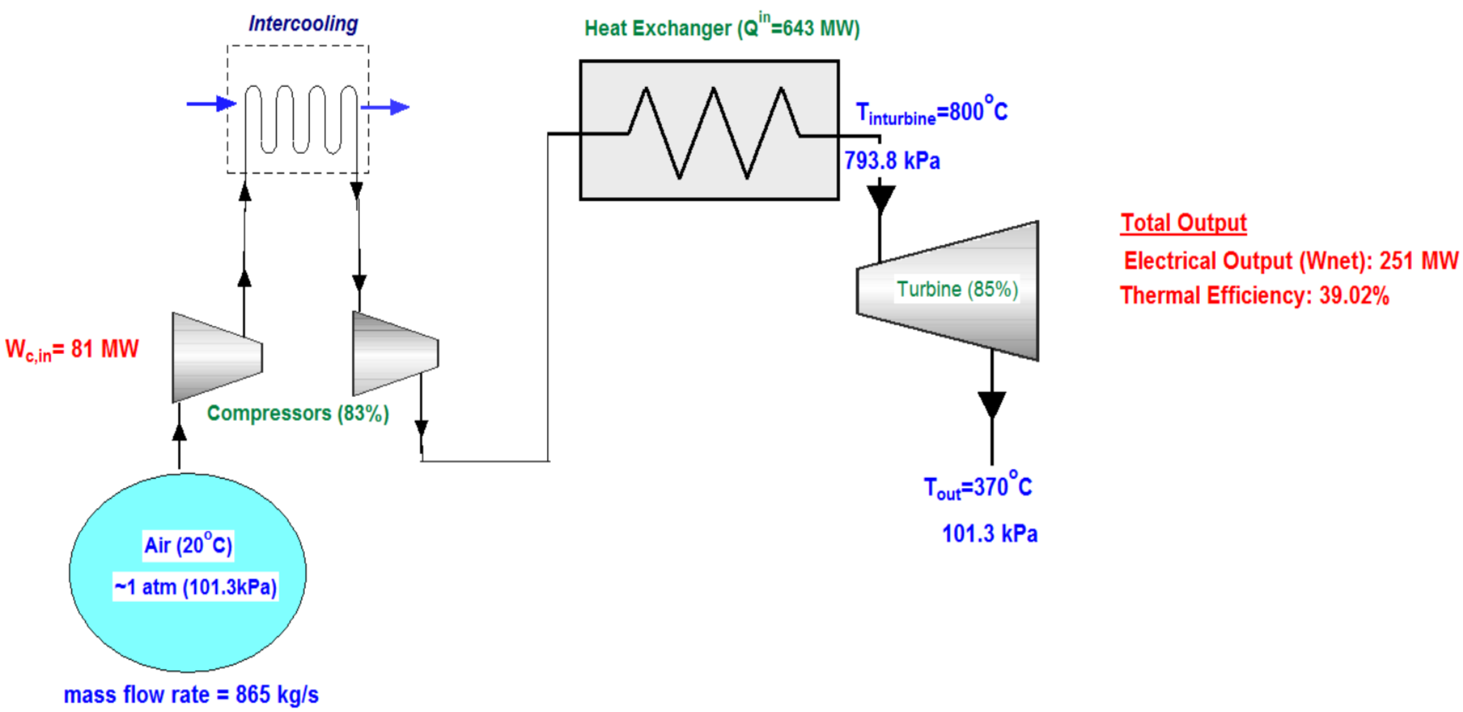

Figure 3.7.1-1: Displays the Air-Flow through the open-air Brayton cycle.

The final open-air Brayton cycle design shows that the compressor work is approximately 81 Megawatts and the turbine work is approximately 412 Megawatts. The Brayton cycle releases the air at 370°C which is sufficiently hot enough to retrieve power from the bottoming Rankine cycle. The electrical output achieved from the Brayton cycle is 251 MW which is right on target with the power output of an original ThorCon reactor which typically operates on a Rankine cycle. The calculation behind the final open-air Brayton cycle can be found in APPENDIX G.

3.6.1.1 Bottoming Rankine Cycle

To maximize power output, the hot air coming out of the Brayton cycle will then be used to convert water into steam and run a bottoming Rankine cycle. This cycle combination type is found most common in natural gas fueled plants in order to increase efficiency. When implemented in our plant, with such a high exhaust temperature, up to 25 percent of the power could be produced by the bottoming Rankine cycle. The design in our plant will have 37% efficiency producing an additional 78 MW of power. This feature will depend on solely on the waste heat from the Brayton cycle and lead to an overall cycle efficiency of 52%. Calculations for these cycles can be found in APPENDIX G and H.

Therefore the overall reactor electrical output of one ThorCon Breeder is 329 MWe which makes the overall nuclear power plant power output of 1.32 GWe which exceeds our goal of 1 GW.

3.6.2 Heat Exchanger

The heat exchanger type chosen for the ThorCon Breeder is a shell and tube design. This type of heat exchanger was chosen because of its capability to extract enormous amounts of heat over its surface area. Using El-Wakil's Log Mean Temperature Difference method, the fluid flow of the fuelsalt and the temperatures required to extract the amount of heat required for our power output was determined. The placement of our heat exchanger is also key in this design. Using the concept from the SM-AHTR design we decided to place our heat exchanger on top of our core to minimize piping and pump work. In doing so that limited the surface are of our heat exchanger to the surface area of our core which is approximately 27.5 m^2.

The fuelsalt inlet and outlet temperatures in the heat exchanger and the mass flow rate of the fuelsalt were based on the thermal power output derived from the Brayton cycle calculations. The mass flow rate of the fuel was found to be 1596 kg/s based on a thermal output of 642 MW and iterations of the inlet and outlet temperatures of the fuelsalt in the heat exchanger.

The inlet temperature was based on the highest temperature of the core which we set as 1000^{o}C. The outlet temperature of the fuel was iterated until the area of the heat exchanger fell within the bounds of the surface area of the reactor core.

In using the LMTD method the MTD difference was iterated and corrected by a correction factor, F, chosen from a plot in *Nuclear Heat Transport,* that gives a correction factor based on the number of tubes and passes and two coefficients, that are based on the inlet and outlet air and fuelsalt temperatures. These coefficients serve as temperature ratios and can be found in equation (8) and (9). For our shell and tube heat exchanger a cross-flow method of the fluid flow will be used which will affect the temperature ratio coefficient equations.

$$P_f = \frac{t_{ao} - t_{ai}}{T_{fi} - t_{ai}} \tag{8}$$

$$R_f = \frac{T_{fi} - T_{fo}}{t_{ao} - t_{ai}} \tag{9}$$

Where P_f and R_f are the temperature ratio coefficients and Tf is the temperature of the fuel and ta is the temperature of the air derived from the Brayton air cycle calculations across the heat exchanger. From the final iteration of the heat exchanger calculation the correction factor was found to be 0.96 from El-Wakil using 10 tubes and 8 passes, and P_f equaling 1.21 and R_f was calculated to be 0.253 which were deemed to be reasonable values compared to the examples in the textbook. With this the corrected mean temperature difference (CMTD) was calculated to be 347.2 which is the temperature difference across the heat exchanger do to the cross-flow of the fluid.

Determining the final surface area and length that would be feasible to fit on top of the reactor was mainly due to the high heat transfer coefficient calculated which was impart due to the high convective heat transfer coefficient of the fuelsalt and the amount of turbulence calculated for both the fuelsalt and air.

In a cross flow system we are able to maximize the amount of heat that can be extracted from the fuelsalt as well. With the final recalculation of the heat exchanger the total height is 2.55 meters with a total surface are of 22.41 m^2 which fits inside the bounds of the surface are of the reactor vessel. Figure 3.6.2-1 shows a fluid flow scheme of the cross-flow of the fluids and table 3.6.2-1 shows the final parameters of the shell and tube heat exchanger.

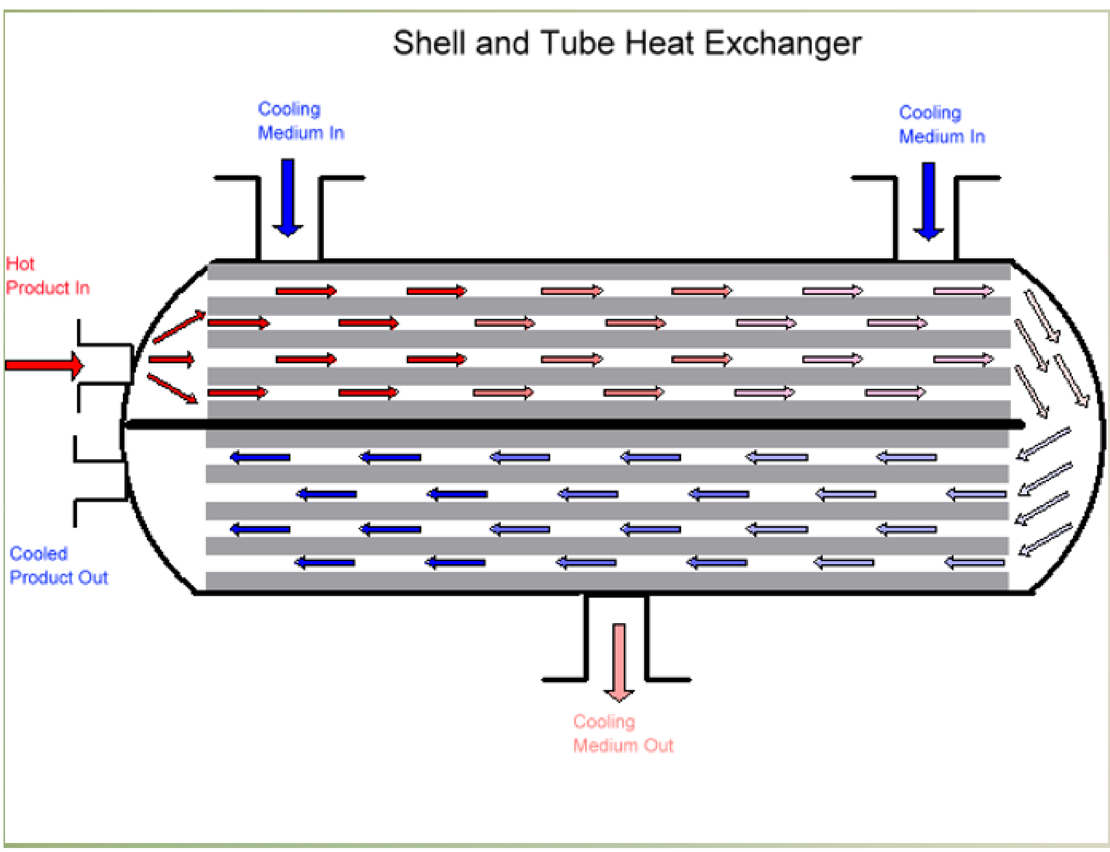

Figure 3.6.2-1: Shell and Tube Heat Exchanger Cross-Flow http://www.genemco.com/aloe/shellandtube.html

Table 3.6.2-1: Heat Exchanger Parameters	
Shell Side Fluid/Diameter (m)	0.035
Tube Side Fluid/Diameter (m)	0.06
Inlet/Outlet Temperature of Air (^oC)	112.8/799.9
Inlet/Outlet Temperature of FuelSalt (^oC)	1000/815
CMTD	347.2
Overall heat transfer coefficient (U)	82611
Reynolds Number of FuelSalt	2.23E05
Reynolds Number of Air	7.414E06
Shell Length (m)	2.54
Shell Arear (m^2)	22.41

3.6.3 Pump Work

As mentioned previously the research of molten salt reactors showed that many designs were able to operate on natural circulation and did not require pump work between the hot and cold fluids of the cooling system. This approach was analyzed for our system. We wanted to implement natural circulation for the fuelsalt to flow upwards through the heat exchanger and back into the core while the air was pushed through, however this proved to not be feasible. Instead two pumps were installed to pump the fuelsalt through the heat exchanger and then back into the core.

The natural circulation method did not work for our design because it required an unrealistic height for the heat exchanger (chimney height). This was shown through the use of equation (10) derived from *Natural Circulation Systems.*

$$\dot{m} = \sqrt[3]{\frac{2\rho_o^2 g\beta QH}{c_p} * \left(\frac{A^2}{\frac{fL}{D}}\right)} \tag{10}$$

Where Q is the total heat generation, H is the chimney height, β is the volume expansivity (buoyancy) coefficient of the fuelsalt, A is the area of the heat exchanger, f is the friction coefficient, and L and D are the length and diameter of the pipes used in the heat exchanger, respectively.

This equation can only be implemented under the following assumptions, that there is a constant heat flux, insulated walls, and a 1-D flow. Our reactor and heat exchanger both meet these assumptions. The heat flux is constant in the core and thus will remain constant in the heat exchanger. The heat exchanger is assumed to be insulated due to high temperature operation and fast heat transfer between the cold air and hot fuelsalt. The flow is also one dimensional being that the flow will only be in the z-direction (vertically). With use of this equation the mass flow rate found for the fuelsalt in the heat exchanger can be used to determine the height, H, needed to attain natural circulation. However, this calculation depended heavily on the volume expansivity coefficient, β, and the total length of pipe, L within the heat exchanger, which made it difficult to attain a realistic height of a heat exchanger that would essentially fit on top of the reactor core.

From numerous iterations of this calculation the required height needed to attain natural circulation was approximately 30,000 meters. Granted this number could have possibly decreased if the surface are of the heat exchanger could be increased. That could not be done because it would make the heat exchanger surface are bigger than the reactors and would not fit inside the pressure vessel. This enormous height was heavily attributed to the amount of fuelsalt we are trying to move and its buoyancy coefficient. The buoyancy coefficient is based on the density changes in the fuelsalt due to temperature which is shown in equation (11).

$$\beta = -\frac{1}{\rho}\left(\frac{d\rho}{dT}\right) \approx -\frac{1}{\rho_{av}} * \frac{\rho-\rho_{av}}{T-T_{av}} \qquad (11)$$

Due to high changes in density of the fuelsalt yielded a very small buoyancy coefficient which in return only increased the height required to attain natural circulation.

The discerning results from the natural circulation calculations did not thwart the design of the ThorCon breeder. Instead of natural circulation 2 pumps were installed into the system. One pump is placed above the reactor in the center of the heat exchanger to pump the salt through the heat exchanger and another is place near the bottom of the reactor to pump the now cool and chemically processed fuel salt back into the reactor core (see figure 3.6.1-1 Fluid Flow Schematic).

Based on the size of the heat exchanger calculated previously using the LMTD method the pump work for both pumps was calculated. Using examples from text books the initial pump work was based on a 60% efficiency, which the class instructor advised was a rather conservative estimate. In order to find the pump work the pressure drop across the heat exchanger and the reactor was needed first.

The pressure drop for the heat exchanger was found using equation (12) and was based on the average density of the fuel salt in each stage.

$$\Delta P = (\rho_{avg} H_{core} - p_{avg} H_{chimney}) \frac{gL}{D} \tag{12}$$

There are many pressure drop equations that could have been used, however they involve many parameters such as viscosity and frictional changes that have not really been studied for the molten salt used in our design. This equation allows us to focus on the density change of the fuelsalt due to temperature changes, which we assumed to be directly proportional for calculation purposes.

The pressure drop in the pipe that takes the salt from the heat exchanger back into the reactor was calculated using equation (13). This equation was used because the friction and radius of the pipe plays a significant part in the pressure gradient in the fluid flow.

$$\Delta P = f \left(\frac{\rho_{pipe}}{2}\right) * \left(\frac{V^2}{l_{pipe}}\right) * D_h \tag{13}$$

Where V is the velocity of the fuelsalt in the pipe which was determined by the mass flow rate and the area of the pipe. The l_{pipe}, was the length of pipe required to move the salt from the top of the heat exchanger to the bottom center of the core. This length equaled the radius of the core plus the height of the core and the height of the heat exchanger, this is approximately 12 meters of pipe.

The pressure drop from each stage was calculated to be 36000 kPa from the reactor core to the heat exchanger and 102 kPa through the pipe. These numbers seem reasonable due to the fact that the reactor is operating near atmospheric pressure so the pipe pressure should be near that and the pressure gradient would be high in the heat exchanger due to the vast change in velocity, temperature, and fluid flow area.

With the calculated pressure drop the work of the pumps were calculated based on the mechanical energy required to overcome the respective pressure drops. Once the mechanical energy was calculated the pump work required was calculated using equation (14).

$$W_{pump} = \dot{E}_{mech}/\eta \tag{14}$$

Where η is the efficiency of the pump.

Through iterations of this calculation and trying to reduce the pump work required to move the fuelsalt. The efficiencies of the pumps were increased to 75% and the inlet and outlet temperatures of the fuelsalt in the heat exchanger were set to 1000^{o}C and 815^{o}C, respectively. With this, the final pump work of the pump in the heat exchanger is 3.3 MW and the pump work required to pump the fuelsalt back into the core is 3.2MW. This seemed fair being that that the same amount of salt is being moved from the center of the core through the heat exchanger and back to the center of the core from the bottom.

Table 3.6.3-1 shows the change in density of the fuelsalt with the change in temperature. Table 3.6.3-2 shows the mechanical energy and work required for each pump based on their respective pressure drops.

Table 3.6.3-1: Density Changes of Fuelsalt		
Temperature (°C)	Density of Fuel Salt (kg/m^3)	
700	2010	ORNL density of fuelsalt
875	1703	Average density in the core
907.5	1656	Average density in the heat exchanger

Table 3.6.3-2: Pump Work Data			
Pumps	Mechanical Energy (kJ)	Mechanical Work (kW)	Pump Work (MW)
Pump 1 (Center of Heat Exchanger)	1.55	2472	3.3
Pump 2 (Bottom of reactor)	1.52	2426	3.2

The calculation that achieved the final pump work can be found in APPENDIX I.

Section 4 – Safety Analysis

The primary objective of this design concerning safety is to analyze two safety design features suggested by ThorCon Power to ensure that in cases where the fuelsalt overheats and potentially damages the reactor vessel that the plant is "walkaway-safe". In other words, the goal is to make sure that in the event of this type of accident, the reactor will react in a way that passively mitigates the consequences of the accident. The primary feature of the reactor design to ensure that the design is 'walkaway-safe' is the inclusion of a freeze-plug between the core and a subcritical containment or fuelsalt drain tank. In the event of overheating, a thermal couple will be triggered to stop the pumps and allow the overheated fuelsalt to drain and melt the freeze-plug and the fuelsalt will evacuate into the drain tank. The drain tank is designed such that the salt will almost immediately go subcritical and will be allowed to cool and give off decay heat which will be removed by the cooling membrane wall. Additionally, our reactor design ensures that the fuelsalt will only be critical within the core. Throughout the rest of the piping (that which goes through the heat exchanger for example), the fuelsalt will be subcritical ensuring that there will not be any heat generation outside of the core, except for the decay heat associated with the fuelsalt. While there is decay heat, the material used for the piping of the reactor is such that the decay heat will not cause any adverse effects.

The second accident scenario is a leak in the breeder blanket material. In this scenario if the all of the thorium breeding material was to leak or the chemical processing were to stop then the cover gas of the entire reactor, argon will fill the breeding section of the reactor until the thorium leak is fixed and pumped back into the reactor. In this we analyzed the criticality of the reactor due to total depletion of the thorium breeder material and the time frame that the reactor can operate at full power if chemical processing were to stop and U-233 can no longer be bred.

This section will also discuss passive and operator safety controls of the ThorCon Breeder that are based on the original ThorCon plant design.

4.1 ThorCon Safety Analysis

4.1.1 Reactivity Feedback in Leakage Scenario

The ThorCon Breeder is designed to fill with a cover gas (Argon) if the thorium breeding material leaks from the system. However, to obey with ALARA rules argon-41 emission must be reduced. According to "Dspace" 86% of the emission of argon-41 comes from the graphite region and has a rather short half-life of about 1.6 hours. To counter this effect however, the graphite region and thorium depletion region should be purged with a cover gas of helium although this would be very costly. Using the method of a graphite helium system is probably the most desired situation. A known experiment using this method would be in reference to MIT's reactor study of reducing emission of argon-41. The graphite helium design consists of a series of reactor grade stingers, and there are many void spaces between around the stingers. These voids could potentially be filled by an influx of air, so an inert helium cover gas blankets the graphite in order to help prevent air from entering the region. The helium is supplied to the graphite region through a constant pressure gasholder at the rate of approximately four cubic feet per hour. The helium is exhausted to the main plenum through the pipe tunnel that runs beneath the reactor [13]. Although ThorCon's design is not the same design as the MITR-II, the same idea applies to both because of the common grounds of graphite.

In the case of the breeding material leaking, the cover gas will fill that space and the reactor will maintain criticality. Running tests, in MCNP6 for the breeding material region filled with argon gas shows negligible change in the criticality of the reactor. This way the reactor is stable so that the thorium can be replenished and any leaks can be fixed. Depending on the severity of the leak the reactor could be shut down or could maintain full power operation for approximately 2 months before the fuel is fully burned up.

4.1.2 Freeze Valve Time Analysis

In order to ensure that the freeze valve could efficiently melt in time to let the overheated fuelsalt drain into the drain tank, a time analysis was conducted on various pipe sizes for the freeze valve. The freeze valve essentially serves as a passive safety mechanism of our system in overheating scenarios. The freeze valve is a cylindrical pipe that is filled with frozen coolant salt. This salt is frozen and maintained at a temperature of 350°C by being actively cooled.

In a case of the reactor core overheating the temperature of the top of the center of the core would need to exceed 1200°C, which is determined based on the stress limits of the reactor vessel material, Hastelloy-N, which begins to stress beyond repair above 1050°C. Implementing a thermal couple at the top of the core to trigger at a temperature of 1150°C (for marginal error of actual temperature) would essentially stop the pumps and allow the overheated fuelsalt to drain due to gravity to the fuelsalt plug. At the same time the actively cooling system will switch to a heating system to help the pipe to warm up and start melting the freeze plug. Once the plug is melted the fuelsalt will drain until all of the fuelsalt is inside of the drain tank. This passive system happens on its own and operators cannot do anything to prevent it which is key.

According to ThorCon it would take approximately 10 minutes for an entire ThorCon reactor to drain. The volume of our fuelsalt region is much less than an original ThorCon design and should therefore take less time. However, the time it takes for our freeze valve to melt is a key element which is what was analyzed.

The time frame required to melt the freeze plug was calculated based on the theory behind *Melting the Cube* by Jones, which basically analyzed the phenomenon of melting an ice cube surrounded by a heating element. This study was adjusted to fit the requirements of our freeze valve. Assuming adiabatic conditions and that the pipe takes on the temperature of the hot fuelsalt the same approach could be applied. The *Melting the Cube* study was based on finding an equation that equates the decrease in the frozen ice volume with respect to time. Our analysis will do the same, relate the volume of the fuelsalt to the time required for the volume to deplete (or melt).

In this, the volume equation was derived for a cylinder rather than a cube, and parameters were altered based on the fuelsalt composition. However the same procedure used in Jones' study was used in this analysis. The volume equation derived for the melting of the volume of the fuelsalt is show in equation (13).

$$V(t) = e^{\frac{ln\left(\pi r_o^2(2r_o+h_o)\right)\rho r L_f - kTt}{r\rho L_f}} - 2\pi r^3 \tag{13}$$

Where:

- V is the volume
- ro and ho is the initial radius and height of the freeze plug respectively
- r is the change in the radius with time
- T is the difference in temperature between the frozen fuelsalt and hot fuelsalt
- t is the time
- ρ is the density of the frozen fuelsalt

From this equation as time increase the volume of the frozen salt plug will decrease. A value near zero for the volume of the salt will represent a completely melted plug. The derivation of equation (13) can be found in APPENDIX K. Using this equation based on different temperature and volumes of salt different time frames of melting were calculated. Table 4.1.2 shows the different timeframes required to melt the frozen salt plug based on the length of the cylindrical freeze valve. In this study it is assumed that it takes approximately 3 minutes to actively heat the pipe from 350°C (Frozen) to 450°C and approximately 6.5 minutes to heat the pipe by 400°C to 750°C. These times are added to give the total time frame to melt the plug based on their respective temperature difference, T.

Table 4.1.2: Time Requirement of Various Melting Parameters and Lengths of Freeze Plug			
Length of Frozen Salt (m)	**Temperature of Surrounding (^oC)**	**Time To Melt (s)**	**Total Time before Draining (min)**
1	750	32	3.53
1	450	168	9.3
1.3	750	34	3.57
1.3	450	173	9.4
1.7	750	42	3.7
1.7	450	178	9.5

This table shows that if the active heating system can heat the pipe efficiently enough to approximately 750^{o}C that the even the longest fuelsalt plug could be melted in essentially 4 minutes. The fast melting of this salt is impart due to its high heat transfer coefficients and easily changed density due to temperature changes. Figure 4.1.2-1 shows a schematic of the freeze plug and draining of the core. This would call for insertions into the Hastelloy-N of our reactor vessel. However, the pipe diameter is 0.1 meters which is almost negligible compared to the size of the entire reactor.

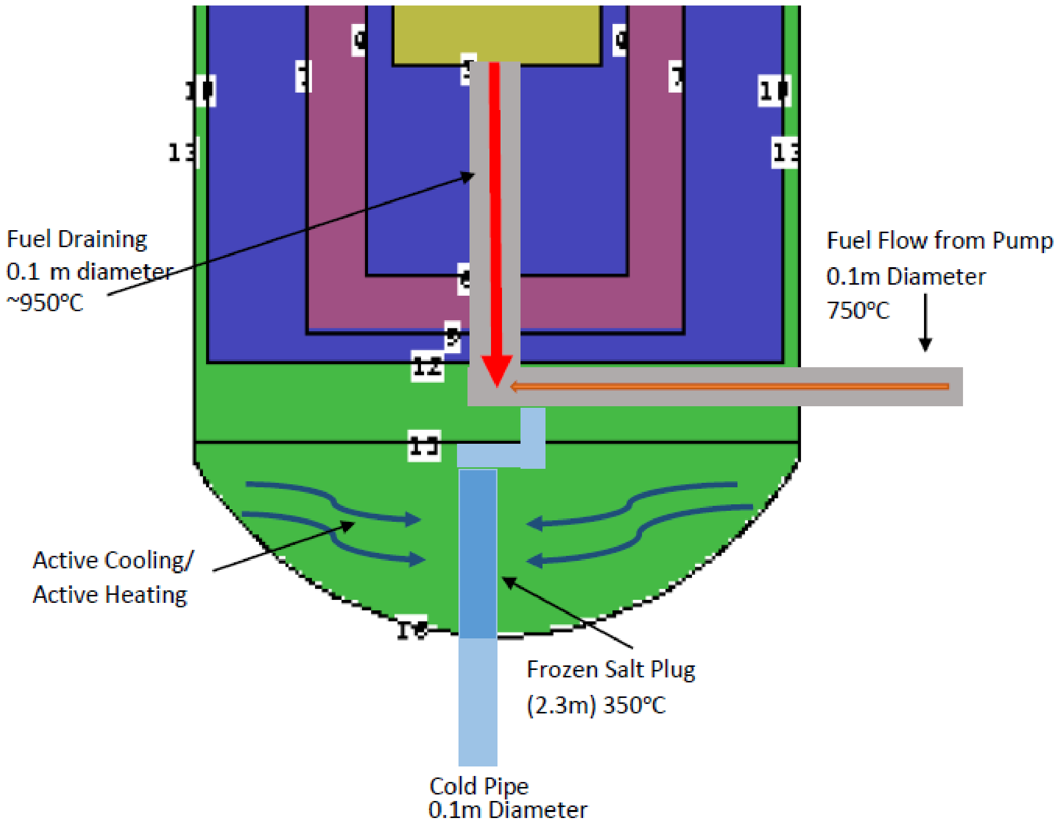

Figure 4.1.2-1: Displays the freeze plug design of the ThorCon Breeder. (Pipes not draw to scale)

4.1.3 Drain Tank Analysis

In the design of the drain tank it was imperative that the fuel salt remained significantly subcritical for the entire duration of draining and cooling. In order to do this we first made assumptions about dimensions and properties of the drain tank. The drain tank must be able to withstand high temperatures without structural failure as it will be used to drain the fuel salt in the case the reactor overheats. Secondly the drain tank must be reasonably short as it will be sitting under the reactor and it would be unreasonable to have it be substantially tall, geometrically. Finally as stated before the salt must remain significantly subcritical to ensure that no heat is generated from fission. The only heat given off from the drain tank should be the decay heat of the fuelsalt.

Next the drain tank was modeled in MCNP keeping the assumptions and design criteria in mind. We chose an interior radius of 1.25 meters because of geometric buckling concerns. To accommodate the volume of fuel salt an interior height of three meters was used. The material used for the walls of the drain tank was 2111 HTR stainless steel. We chose this material because it can resist oxidation and material failure at continuous temperatures of 1150°C and spikes of temperatures higher than that.[1]

A model was made in MCNP and a k-code analysis was used to measure the K_{eff} of the fuel salt at both 1150 ºC and 750 ºC. An initial analysis in MCNP gave a k_{eff} of 1.16, significantly too high. This means that a neutron absorber had to be used. Different neutron absorbing geometries were tested using boron carbide as the absorbing material. The final design settled on was a central column two meters tall with a radius of five centimeters. Figure 4.1.3-1 shows a cross section of the drain tank.

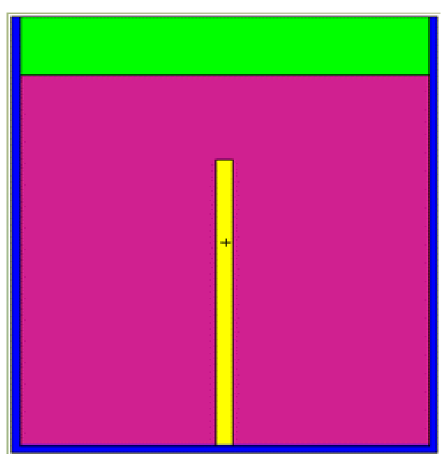

Figure-3.1.1.3-1: A cross section of the drain tank with 1150 ºC fuel salt, the blue region represents 2111 HTR stainless steel, the pink region is fuel salt, the yellow region is boron carbine, and the green region is simply cover gas.

A Kcode of the final design gave a K_{eff} of 0.804 for a temperature of 1150ºC and a K_{eff} of 0.847 for a temperature of 750ºC. This was more than subcritical enough to operate reasonably.

4.2 Overall Plant Safety

As aforementioned the ThorCon reactor liquid fuel design is made such that if the reactor overheats for whatever reason, the reactor will automatically shut itself down. The primary fuelsalt loop will drain from the core into the drain tank and passively handle the decay heat [6]. There is no need for operators to intervene especially since once this happens there is no way to prevent it.

The ThorCon Breeder plant will be built as the original ThorCon reactor plant. The ThorCon Breeders will be 30 m underground with four gas tight barriers between the fuelsalt and the atmosphere 6. The ThorCon Breeder is also designed to operate near atmospheric pressure. This is done so that in the event of a primary loop leak or rupture there is no dispersal energy or phase change of the fluids. Any spilled fuel or breeder material flows to the drain tank where it is cooled. In the case of draining the fuel salt, troublesome fission products are chemically bound to the fuelsalt and will go to the drain tank as well to decay [6].

Another great feature of the ThorCon design is that no complex repairs are attempted on site. Its modular stature allows for every aspect of the ThorCon plant to be replaceable with little to no interruption in power output. Even the spent fuel that is drained to the decay tank in overheating cases can be shipped to a Centralized Recycling Facility (CRF). After 4 years the spent fuel decay heat will be down to 0.25% of its original operating decay heat and can be transferred to CRF.

4.3 Things to Consider in the ThorCon Breeder Design

The ThorCon Breeder is designed to operate under the assumption that the fuelsalt remains turbulent in the core. This turbulences helps with the transfer of heat, fuelsalt flow in the heat exchanger, and minimization of hotspots in the core. One thing to consider to ensure turbulence is the addition of fins to the inside of the graphite moderator that can help produce turbulence in the fuel salt as it is being pumped through the core. These fins would need to be thick enough to withstand the pressure of the incoming fuelsalt and the high temperatures. Graphite or a strong moderating medium could possibly meet this requirement.

The cooling system of the ThorCon breeder is designed to operate on a single loop of heat transfer from the fuelsalt to the air. This method was advised by course instructors as bold but doable. Another method to consider is the implementation of ThorCon's original design of four loop separation of heat from the fuelsalt. This method involve a primary fuelsalt loop, secondary fluoride salt loop, a tertiary solar salt loop, and a final high pressure steam loop. Implementing this could eliminate the possibility of fuelsalt leaking into the air and causing contamination if there was a primary loop rupture. If this was to happen the excess air sent to the turbine could not be released into the atmosphere and operation would have to be stopped immediately.

Section 5 – Economics & Fuel Cycle Cost Analysis

Due to the importance of economic considerations related to a new reactor, an estimated fuel cycle cost was calculated. This fuel-cycle cost model is based on the 4-year lifetime of a single core. The costs related to as single core can then be iterated as many times as necessary for the projected lifetime of the plant as a whole.

The costs associated with the construction of a single core will be brought to core-startup. These costs include the fabrication of the coolant salt, the containment material, as well as the fuel/breeder salts. At beginning of life, the reactor core needs to be doped initially with Uranium-235. The necessary amount of fissile material that needs to be included in the reactor core prior to the establishment of breeder equilibrium is 3520 kg. The costs associated with the enrichment and fabrication of the startup material are also brought to core-startup.

5.1 - Cost of a Single Can

For ThorCon, the cost of a Can is a consumable. Under this model, all materials explicitly associated with the can are taken into consideration for cost analysis. These materials include the heat exchanger, the skeleton for the fuelsalt core and breeding blanket, as well as the freeze-valve and drain tank. The total costs associated with production of a single can are found in Table 6.1.

Table 6.1 – Cost of a Single LOKI Reactor Can

Section	Cost ($)
Skeleton	7339451
Heat Exchanger	230000
Freeze Plug	907.32
Drain Tank	231721.4
Total	7802079.72

5.1.1 - Heat Exchanger Cost Analysis

The LOKI design utilizes a shell-and-tube heat exchanger with the following properties:

Table 5.1.1 - Geometric Properties of A Single LOKI Heat Exchanger

Heat Exchanger Area	21.9 m^2
Length/Diameter of Tubes	13.94 m/0.05 m
Shell Length	3.92 m

Following these design parameters, the cost of a single heat exchanger is estimated at $230000.

5.1.2 - Freeze Plug Cost Analysis

The LOKI freeze plug is a 0.1 meter diameter by 2.3 meter length cylinder constructed out of the same material as the coolant salt, $NaF\text{-}BeF_2$. These dimensions correspond to a volume of 0.018 m3. The operating density of the freeze-plug is 5588 kg/m^3. Therefore, the total mass of a single freeze plug is approximately 100 kg. The molar mass of $NaF\text{-}BeF_2$ is approximately 89 g/mol. Therefore, there are 1123 moles of the coolant salt in the 100kg freeze plug. This corresponds to a BeF_2 mass of 52.921 kg and a NaF mass of 47.078 kg. Table 5.1.2 summarizes the results of the freeze plug cost analysis.

Table 5.1.2 - Cost Analysis of Freeze Plug

Material	Mass (kg)	Cost ($) /kg	Cost ($)
NaF	47.078	0.5	23.539
BeF_2	52.921	16.7	883.78
Total	100	9.07	907.32

5.1.3 - Can Skeleton Cost Analysis

The "skeleton" of the can is defined as all of the materials included in the can that are not part of the heat exchanger, freeze valve, or drain tank. This includes all of the graphite, hastelloy, and various odds-and-ends such as heating tape and aerogel insulation. The cost breakdown of the skeleton can be found in Table 6.1.2.

Table 5.1.3 - Skeleton Cost of a Single LOKI Can

Material	Amount	Unit-Cost ($)	Cost ($)
tzm (kg)*	529	50	26450
graphite(kg)	169551	20	3391020
Hastelloy (kg)	3.04E+05	10.293	3.13E+06
PLP Pump (kW)	6530	75	489750
Heating Tape (m^3)*	170	1600	272000
Aerogel Insulation (m^2)*	170	63	26456
		Can Total	7,339,451.901

(*) Indicates quantities shared between the LOKI and ThorCon design.

5.1.4 - Drain Tank Cost Analysis

The materials and associated cost of production for the LOKI drain tank may be found in Table 5.1.4.

Table 5.1.4 - Estimated Cost of One LOKI Drain Tank

Material	Quantity	Unit-Cost ($)	Cost ($)
2111 htr stainless (kg)	13514.2	17	229741.4
Boron Carbide (kg)	39.6	50	1980
Total ($)			231721.4

5.2 - Fuel Cycle Cost

Due to the breeding capability of the LOKI design, the only major fuel-cycle costs associated with the design are those associated with the enrichment of Uranium-235 necessary for initial operation. Once the breeding cycle is established, all further fuel production is performed in-house and included in standard operation & maintenance. However, continued breeding capability requires an appreciable inventory of Thorium salt. The cost analysis for the production of the necessary fuel and breeder salts for a 4-year operating cycle are included in table 5.2.

Table 5.2 - LOKI Fuel Cycle Costs

Material	Amount	Unit-Cost ($)	Cost ($)
ThF_4 (kg)	1.00E+04	14.4	1.44E+05
NaF (kg)	1.07E+04	0.5	5.36E+03
BeF_2 (kg)	1.77E+03	16.7	2.95E+04
	Total	31.6	1.79E+05
Uranium Fuelsalt			

Material	Amount	Unit-Cost ($)	Cost ($)
UF4 (kg)	3.52E+03	3114.72368	1.10E+07
NaF (kg)	1.61E+04	0.5	8.04E+03
BeF2 (kg)	2.84E+03	16.7	4.75E+04
Total		**3131.92368**	**1.10E+07**
Uranium Enrichment Process Costs			
Discount Rate	5	Capacity Factor	1
Fuel U235 Percent	20	Feed U3O8 %	0.72
Tails U235 Percent	0.2	Feed Kg/ Fuel kg	38.08
U3O8 $/kg	81.57	U3O8 $ / kg	81.571
USD/SWU	90	U3O8 to UF6 $/kg	7.5
		UF6 to UF4 $/kg	1
U3O8 USD per kg U235	18313.92		
Conversion USD per kg U235	1427.88		
SWU USD per kg U235	20417.37		
U235 USD per kg	40159.173		

5.3 – Staff Costs

A preliminary estimate on the staff costs for operation of a 1-GWe LOKI reactor is informed by analysis performed by ThorCon. Considering the vast similarities between the two designs, it is reasonable to conclude that the total employment of the LOKI design will be identical to that of the ThorCon design. This estimate can be found in Table 6.3.

Table 6.3 – On-Site Staff for 1-GWe LOKI Plant

Administration and Training	**25**
Operations	**42**
Maintenance	**30**

Janitorial, grounds, cafeteria	**10**
Security, Loss Prevention	**72**
Engineering, work control	**10**
Trainees	**20**
Total	**209**

The true nature of annual payroll will be heavily dependent on where the plant is located. Without such information, a conservative estimate of $150,000 per man-year is used. This corresponds to a levelized cost of approximately 0.5 cents per kWh for a 1-GWe plant operating with a capacity factor of 0.9 [4].

5.4 – Total Costs

Consolidating the above analysis, a final estimate of total plant caused is made. The results contained in Table 6.4.

Table 5.4 – Total Cost Estimate of 1-GWe LOKI Plant

Plant Electrical Power (kWe)	1000
Plant life (years)	32
Construction Period (years)	4
Capacity Factor	0.90
Cost of capital	0.10

Overnight Cost	$800 MM	$1000 MM	$1200 MM
Unit Capex	0.01236	0.01545	0.01845
Unit Can Cost	0.231	0.231	0.231
Unit Fuel Cost	0.00511	0.00511	0.0511
Unit Salt Cost	0.00020	0.00020	0.00020
Unit staff	0.00493	0.00493	0.00493
Unit waste	0.00100	0.00100	0.00100
Total $/kWe	0.02699	0.03008	0.03317

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APPENDIX A

Buckling Calculations

Material Composition			
NaFBeF2			
Rho (g/cm3)	Molar Mass (g/mol)	mol/cm^3	#/cm^3
2.01	38.9	0.051670951	3.11162E+22

Neutronic Properties								
Na	**oa**	**os**	**Ea**	**Es**	**Ef**	**Et**	**Etr**	**muo**
3.1112E+22	5.30E-01	1.66	1.65E-02	0.05165297	0	6.81E-02	6.66E-02	0.029
Be	oa	os	Ea	Es	Ef	Et	Etr	muo
3.1112E+22	0.01	7	3.11E-04	0.217813728	0	2.18E-01	2.02E-01	0.0741
F	oa	os	Ea	Es	Ef	Et	Etr	muo
3.1112E+22	0.0096	4.018	2.99E-04	0.12502508	0	1.25E-01	1.21E-01	0.0351
F2	oa	os	Ea	Es	Ef	Et	Etr	muo
6.2232E+22	0.0096	4.018	0.000597432	0.25000159	0	0.250648	0.241252429	0.0351
F4	oa	os	Ea	Es	Ef	Et	Etr	muo
1.2446E+23	0.0096	4.018	0.001194864	0.50010318	0	0.501295	0.482504857	0.0351
			Ea	Es	Ef	Et	Etr	
Total for NaF			1.68E-02	0.17668049	0	1.93E-01	1.88E-01	
Total for BeF2		9.09E-04	0.467863887	0	4.69E-01	4.43E-01	totalmuo	0.2084

Material Composition			
UF4			
Rho (g/cm3)	Molar Mass (g/mol)	mol/cm^3	#/cm^3
6.7	314.022	0.021336085	1.28486E+22

Neutronic Properties									
U233									
#/cm^3	oa	os	Ea	Es	Ef	Et	of	muo	Etr
9.62359E+21	2.68E+00	8.871	2.58E-02	0.085370903	5.05238689	1.11E-01	525	0.002	1.11E-01
Th-232									
#/cm^3	oa	os	Ea	Es	Ef	Et	of	muo	Etr
3.225E+21	7.37E+00	13.36	2.38E-02	0.043085949	0.02438097	6.69E-02	7.56	0.0028	6.67E-02
F4	oa	os	Ea	Es	Ef	Et	Etr	muo	

5.13944E +22	0.0096	4.0 18	0.000493 386	0.206502 542	0	0.20699 593	0.199236 985	0.0351	
			Ea	Es	Ef	Et		totalm uo	Etr
Total Thf4			2.43E-02	0.249588 491	0	2.74E-01			2.66 E-01
Total UF4			2.63E-02	0.291873 445	5.05238 689	3.18E-01			3.10 E-01

Neutronic Properties									
Graphite									
#/cm^3	oa	os	Ea	Es	Ef	Et	Etr	muo	
8.03E+22	4.00E-03	4.8	3.21E-04	0.38544	0	3.86E-01	3.64E-01		0.0556
		Total	3.21E-04	0.38544	0	3.86E-01	3.64E-01		0.0556

Geometrical buckling

Bg^2
Equation used:
(pi/H)^2+(2.405/R)^2

vo	**pi**
0.25	3.141592654
H (m)	**R (m)**
6.95	2.90
H (cm)	**R (cm)**
695	290

Goal Bg^2
0.27744

Material Buckling

Bm^2
(nu*Sigmaf-Sigmaa)/D
D = 1/(3(Etr-muo*Es))
Nu=2.3

Team LOKI

Constituent	% makeup	Ea	Es	Ef	Et	Etr	k
NaF	76.00%	1.28E-02	0.134275317	0	1.47E-01	1.43E-01	1.017728361
BeF2	12.00%	1.09E-04	0.056143666	0	5.63E-02	5.32E-02	
ThF4	9.80%	2.38E-03	0.024459672	0	2.68E-02	2.61E-02	
UF4	2.20%	5.78E-04	0.006421216	0.111152512	7.00E-03	6.83E-03	
Total		1.58E-02	0.221299872	0.111152512	2.37E-01	2.29E-01	

D (cm)
133882413

Material Buckling percentages

Materials	#/bcm	percentage	Column1
Na	0.023648348	0.330871937	3.31E-01
Be	0.023648348	0.330871937	3.31E-01
F	0.023648348	0.330871937	3.31E-01
U238	0.000211719	0.002962232	2.96E-03
Th233	0.000316050	0.004421956	4.42E-03
Total	0.071472811		

Bm^2
0.25219780

APPENDIX B

Breeding Ratio MCNP CODE

```
*-mcnpgen-*- group6 u-233/thorium,
c SYSTEM WITH Control rods and reflector (no vessel)BR
c
c    cell cards
1    4   -2.01  -1 -2 3 imp:n=1 $ homog core (fuel,salt)
2    2   -2.23      (-4 -5 6) (1:2:-3) imp:n=1 $ graphite moderator
3    1   -2.5   (-7 -8 9) (4:5:-6) imp:n=1 $ breeder blanket
4    2   -2.23  (-10 -11 12) (7:8:-9) imp:n=1 $ graphite reflector
5    0       #1 #2 #3 #4 imp:n=0 $ outside world

c    surface cards
1    cz  100  $ outer radius of core
2    pz  285.5   $ top of core
3    pz -50.5   $ bottom of core
c
4    cz  142  $ outer radius of cyl graphite reflector
5    pz  285.5 $ top of graphite reflector
6    pz -235.5$ bottom of graphite reflector
c
7    cz  180 $ radius of blanket
8    pz  285.5 $ top of blanket
9    pz -285.5 $ bottom of blanket
c
10   cz  275 $ graphite moderator
11   pz  285.5 $ top graphite
12   pz  -310.5 $ bottom graphite

c   material cards
mode n $ transport neutrons
c
m1    90232.70c  0.02227 $Th
      9019.70c   0.08909 $F
      4009.70c   0.04029 $Be
      9019.70c   0.08057  $Flouride
      11023.70c  0.3837  $Na
      9019.70c   0.3837  $Flouride
m2    6000.70c   1      $graphite
m4    92233.70c  0.001  $Uranium 233
      92238.70c  0.004  $U-238
      9019.70c   0.02         $Flouride
      4009.70c   0.0443 $Be
      9019.70c   0.08867  $Flouride
      11023.70c  0.421    $Na
       9019.70c  0.421    $Flouride
m3    5010.70c   0.0995   $Boron
      5011.70c   0.4005
              6000.70c   0.49 $Carbon
c
```

```
m5 92233.70c   1 $ pure u233 for tally
c
m6 90232.70c   1 $ pure th for tally
c
print 40 $ print material card information
c
c Source Cards
c
c tallies
c
fc4 neutron flux in cell
f4:n 1 3
e4 0.625e-6 0.1 5
fm4 5.52e19 $norm(nuetron/sec) 6.4e8watt*2.5neut/fiss*1fiss/180.88MeV*1MeV/1.602e-13
wattsec=5.52e19 nuet/sec
fq4 f e
c
fc14 u233 (fission/scrneutron)
f14:n 1
sd14 1
fm14 6.0145e-5 5 -6
c
fc34 th232 gamma rate (production of U233,reactions/srcneutron) in cell 3
f34:n 3
sd34 1
fm34 1.68078e-2 6 102
c
c    Criticality Control Cards
kcode 1000 1 50 250
ksrc   0 0 0
      0 0 10
      0 0 20
      0 0 30
      0 0 40
      0 0 50
      0 0 60
      0 0 70
      0 0 80
      0 0 90
      0 0 100
      0 0 110
      0 0 120
      0 0 130
      0 0 140
      0 0 150
      0 0 160
      0 0 170
      0 0 180
```

```
      0 0 190
      0 0 200
      0 0 210
      0 0 220
      0 0 230
      0 0 240
      0 0 250
      0 0 260
      0 0 270
      0 0 280
               0 0 284
        200 0 0
         -200 0 0
          0 200 0
          0 -200 0 $ runs with added ksrc points
c
```

APPENDIX C

Reactor Burn Up MCNP Code

```
*-mcnpgen-*- group6 u-233/thorium,
c SYSTEM WITH Control rods and reflector Burn up
c
c    cell cards
1    4   -2.03  -1 -2 3 imp:n=1 $ homog core (fuel,salt)
2    2   -2.23      (-4 -5 6) (1:2:-3) imp:n=1 $ graphite moderator
3    1   -2.5   (-7 -8 9) (4:5:-6) imp:n=1 $ breeder blanket
4    2   -2.23  (-10 -11 12) (7:8:-9) imp:n=1 $ graphite reflector
5    0       #1 #2 #3 #4 imp:n=0 $ outside world

c    surface cards
1    cz  100  $ outer radius of core
2    pz  285.5   $ top of core
3    pz -50.5   $ bottom of core
c
4    cz  125  $ outer radius of cyl graphite reflector
5    pz  285.5 $ top of graphite reflector
6    pz -235.5$ bottom of graphite reflector
c
7    cz  180 $ radius of blanket
8    pz  285.5 $ top of blanket
9    pz -285.5 $ bottom of blanket
c
10   cz  275 $ graphite moderator
11   pz  285.5 $ top graphite
12   pz  -310.5 $ bottom graphite

c   material cards
mode n $ transport neutrons
c
vol 2.24838E+07 4j $ need to provide volume for cell 1 for burn card
c
c    Criticality Control Cards
kcode 1000 1 100 400
ksrc   0 0 0
     0 0 10
     0 0 20
     0 0 30
     0 0 40
     0 0 50
     0 0 60
     0 0 70
     0 0 80
     0 0 90
     0 0 100
     0 0 110
     0 0 120
     0 0 130
```

```
      0 0 140
      0 0 150
      0 0 160
      0 0 170
      0 0 180
      0 0 190
      0 0 200
      0 0 210
      0 0 220
      0 0 230
      0 0 240
      0 0 250
      0 0 260
      0 0 270
      0 0 280
               0 0 284
        200 0 0
         -200 0 0
          0 200 0
          0 -200 0 $ runs with added ksrc points
c
c
c Burnup Calculations (burn card must be below kcode card and above material card)
burn time=90 90
    power=642
    pfrac=1.0 1.0
    mat=4
    matvol=2.24838E+07
            omit=4 9    7016 8018 8019
                    9018 10021 10022
                                    12023 13026 91230
    bopt=1 4 -1 $ (q=1,0=tier1 fission product+4=print output at end,-1=models off)
c
m1    90232.70c  0.02227 $Th
      9019.70c   0.08909 $F
      4009.70c   0.04029 $Be
      9019.70c   0.08057  $Flouride
      11023.70c  0.3837  $Na
      9019.70c   0.3837  $Flouride
m2    6000.70c   1      $graphite
m4    92233.70c  0.001  $Uranium 233
      92238.70c  0.004  $U-238
      9019.70c   0.6077 $Flouride added fractions
      4009.70c   0.0443 $Be
      11023.70c  0.421    $Na
m3    5010.70c   0.0995   $Boron
      5011.70c   0.4005
              6000.70c   0.49 $Carbon
```

```
c
c
print 40 $ print material card information
c
c Source Cards
c
c tallies
c
fc4 neutron flux in cell
f4:n 1
e4 0.625e-6 0.1 20
fm4 5.52e19 $norm(nuetron/sec) 6.4e8watt*2.5neut/fiss*1fiss/180.88MeV*1MeV/1.602e-13
wattsec=5.52e19 nuet/sec
fq4 f e
c
```

APPENDIX D

ThorCon Breeder Reactor MCNP Code

```
*-mcnpgen-*- group6 u-233/thorium,
c SYSTEM WITH Control rods and reflector and vessel N-FLUX
c
c    cell cards
1    4   -2.13  -1 -2 3 imp:n=1 $ homog core (fuel,salt)
2    2   -2.23 (-4 -5 6) (1:2:-3) imp:n=1 $ graphite moderator
3    1   -2.5   (-7 -8 9) (4:5:-6) imp:n=1 $ breeder blanket
4    2   -2.23  (-10 -11 12) (7:8:-9) imp:n=1 $ graphite reflector
5    5   -8.86  (-13 -14 15) (10:11:-12) imp:n=1 $ reactor vesssel
6    5  -8.86   (-16 -15)-13 imp:n=1
7   0       #1 #2 #3 #4 #5 #6 imp:n=0 $ outside world

c    surface cards
1     cz  100  $ outer radius of core
2     pz  285.5   $ top of core
3     pz -50.5   $ bottom of core
c
4     cz  125  $ outer radius of cyl graphite reflector
5     pz  285.5 $ top of graphite reflector
6     pz -235.5$ bottom of graphite reflector
c
7     cz  180 $ radius of blanket
8     pz  285.5 $ top of blanket
9     pz -285.5 $ bottom of blanket
c
10    cz  190 $ graphite moderator
11    pz  285.5 $ top graphite
12    pz  -310.5 $ bottom graphite
c
13    cz  290 $hastelloy vessel
14    pz  285.5 $top of vessel
15    pz  -380.5 $bottom of vessel
16    sz  -200.5 350

c   material cards
mode n $ transport neutrons
c
m1    90232.70c  0.02227 $Th
      9019.70c   0.08909 $F
      4009.70c   0.04029 $Be
      9019.70c   0.08057  $Flouride
      11023.70c  0.3837  $Na
      9019.70c   0.3837  $Flouride
m2    6000.70c   1      $graphite
m4    92233.70c  0.001  $Uranium 233
      92238.70c  0.004  $U-238
      9019.70c   0.02         $Flouride
      4009.70c   0.0443 $Be
```

```
      9019.70c   0.08867  $Flouride
      11023.70c  0.421    $Na
       9019.70c  0.421    $Flouride
m5    28059.70c      .723324 $Ni
      24052.70c     .0713136 $Chromium
               42095.70c     .163003 $Molybednum
               26056.70c      .0101877 $Iron
               14028.70c        .0101877 $silicon
               12024.70c        .00815013 $mg
               6012.50c                    .000085013 $carbon
               27059.70c        .00203753 $co
               29063.60c        .00356568 $cu
               74184.70c        .00509383 $W
               13027.70c        .00152815 $Al
      22047.70c      .00152815 $Ti
c
m8 92233.70c   1 $ pure u233 for tally
c
m6 90232.70c   1 $ pure th for tally
c
print 40 $ print material card information
c
c
c Source Cards
c
c tallies
c
fc4 neutron flux in cell
f4:n 1
e4 0.625e-6 0.1 20
fm4 5.52e19 $norm(nuetron/sec) 6.4e8watt*2.5neut/fiss*1fiss/180.88MeV*1MeV/1.602e-13
wattsec=5.52e19 nuet/sec
fq4 f e
c number of fission reactions in core units of fission/sec-cm3
f14:n 1 $cell 1
fm14 6.014E-5 8 -6
c
c mesh tally flux
fmesh134:n geom=cyl origin=0,0,-380.5 axs=0,0,1 vec=1,0,0
    imesh=30 60 90 120 150 180 210 240 270 300
             iints=10 10 10 10 10 10 10 10 10 10
             jmesh=666 jints=222
    kmesh=1.0 kints=1
fm134 5.52e19
c
c    Criticality Control Cards
kcode 1000 1 50 500
ksrc   0 0 0
```

```
      0 0 10
      0 0 20
      0 0 30
      0 0 40
      0 0 50
      0 0 60
      0 0 70
      0 0 80
      0 0 90
      0 0 100
      0 0 110
      0 0 120
      0 0 130
      0 0 140
      0 0 150
      0 0 160
      0 0 170
      0 0 180
      0 0 190
      0 0 200
      0 0 210
      0 0 220
      0 0 230
      0 0 240
      0 0 250
      0 0 260
      0 0 270
      0 0 280
              0 0 284
        200 0 0
         -200 0 0
          0 200 0
          0 -200 0 $ runs with added ksrc points
c
```

APPENDIX E

ThorCon Breeder Homogenized MCNP Code

```
c     cell cards
1     4 -18.95 -1 -2 3  imp:n=1 $ homog core (fuel,salt)
2     2 -2.23  (-4 -5 6) (1:2:-3) imp:n=1 $ graphite moderator
3     1 -11.27  (-7 -8 9) (4:5:-6) imp:n=1 $ breeder blanket
4     2 -2.23  (-10 -11 12) (7:8:-9) imp:n=1 $ graphite reflector
5     0 #1 #2 #3 #4 imp:n=0 $ outside world

c     surface cards
1     cz  10    $ outer radius of core
2     pz  285.5   $ top of core
3     pz -209.5   $ bottom of core
c
4     cz  190  $ outer radius of cyl graphite reflector
5     pz  285.5 $ top of graphite reflector
6     pz -235.5$ bottom of graphite reflector
c
7     cz  205 $radius of blanket
8     pz  285.5 $top of blanket
9     pz  -285.5 $bottom of blanket
c
10    cz  275 $graphite moderator
11    pz  285.5 $top graphite
12    pz  -310.5 $bottom graphite

c     data cards
mode n p $ transport neutrons and photons
c
m1     90232.70c  0.02228 $Th
       9019.50c   0.08912 $F
       4009.50c   0.0404  $Be
       9019.50c   0.0808  $Flouride
       11023.50c  0.3837  $Na
       9019.50c   0.3837  $Flouride
m2      6012.50c  1       $graphite
m4     92233.70c 0.0005   $Uranium 233
       9019.50c  0.002         $Flouride
       4009.50c  0.0443       $Be
       9019.50c  0.08867  $Flouride
       11023.50c 0.421    $Na
        9019.50c 0.421    $Flouride
c
c tallies
c     Criticality Control Cards
kcode 5000 1.0 75 250
ksrc    0.0 0.0 -100.0
        0.0 0.0 0.0
        0.0 0.0 100.0 $ 3 initial source locations
```

APPENDIX F

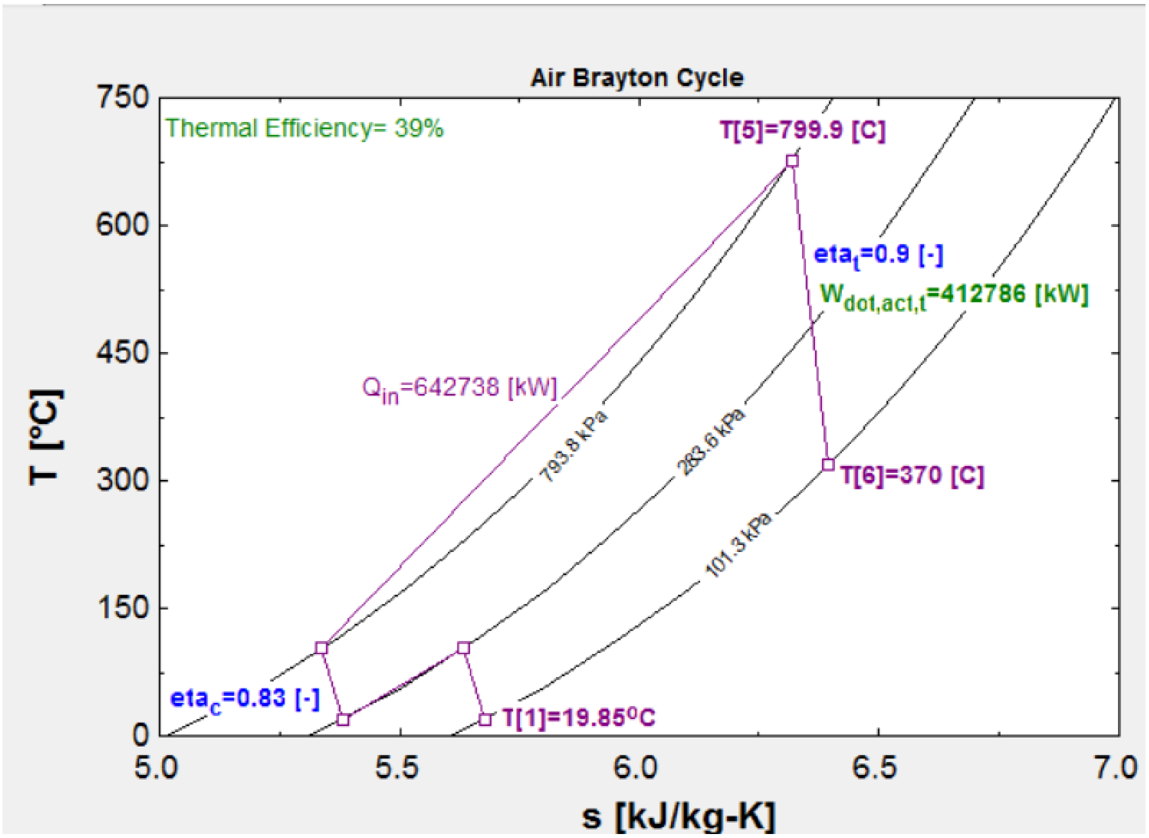
Drain Tank Analysis MCNP Code

```
Drain Tank Analysis
C Cell Cards begin
1 1 -1.78 -1 4 -2 imp:n=1
2 2 -7.8 -3 1 4 imp:n=1
3 3  -2.52 -4 imp:n=1
4 4 -.001784 -1 2 imp:n=1
100 0 3 imp:n=0
C Cell Cards end then blank line

C Surface Cards begin
1 RCC 0 0 0 0 0 300 125
2 PZ 259.45
3 RCC 0 0 -5 0 0 310 130
4 RCC 0 0 0 0 0 200 5
C Surface Cards end then blank line

C Data Cards begin
M1    92233.70c 0.001    $Uranium 233
      92238.70c 0.004
      9019.50c  0.6077    $Flouride
      4009.50c  0.0443    $Be
      11023.50c 0.421    $Na
M2    26056.70c .6568     $2111HTR SS
     6000.70c .0005
     14028.70c .011
     24052.70c .20
     28058.70c .10
     7014.70c .0014
     58140.70c .0003
M3    5010 .8
     5011 3.2
     6012 1
M4    18040 1
KCODE 20000 0.98 50 125
ksrc    0 10 50
Fmesh4:n geom=cyl origin=0 0 -2 axs= 0 0 1 vec=0 1 0
      imesh=10 20 30 40 50 60 70 80 90 100 110 120 130
      iints=5 5 5 5 5 5 5 5 5 5 5 5 5
      jmesh=30 60 90 120 150 180 210 240 270 300
      jints=5 5 5 5 5 5 5 5 5 5
      kmesh=1.0
      kints=1

Eveything after this point is ignored by MCNP
```

Volume fuelfalt=1.055e7
mass fuelsalt=22640 kg
density at 750C 2.01
Density at 110 1.78
volume 750 = 1.126e7
volume 1100=1.272e7
cross section at bottom (*control rod region)=49008
volume control rod region=9.801e6 cc
cross section next region=49087.4 cm
height of fuelsalt 1100=259.45
height of fuelsalt 750=229.71
K_1100=0.80400
k_750=0.84687

APPENDIX G

Open Air Brayton Cycle




```
"OPEN AIR BRAYTON CYCLE-VNwadeyi"
"Constants"
R$= 'air'                                   "material used"
cp_air=1.014 [kJ/kg-K]                      "specific heat of air"
m_dot= 865 [kg/s]                           "mass flow rate"
P[1]=1 [atm]*convert(atm,kPa)               "inlet pressure of 1st compressor"
T[1]=converttemp(K,C,293[K])                "Inlet room temperature of air"
T[5]=converttemp(K,C,1073[K])               "inlet temperature of turbine"
k=1.40                                      "specific heat ratio"
rp = (1073/293)^(k/(2*(k-1)))               "pressure ratio ASSUMED"
q_dot_coreth=557 [MW]*convert(MW,kW)        "Thermal Output ThorCon"
eta_c=0.83 [-]                              "compressor effeciency"
eta_t=0.9 [-]                               "turbine effeciency"
v=1.29 [m^3/kg]                             "specific volume"

"State1"
s[1]=entropy(R$,T=T[1] ,P=P[1])
h[1]=enthalpy(R$, T=T[1])
V[1]=volume(R$,T=T[1],P=P[1])

"State2"
P[2]=(rp^.5)*P[1]
s_s[2]=s[1]
h_s[2]=enthalpy(R$,s=s_s[2],P=P[2])
W_dot_s_c=m_dot*(h_s[2]-h[1])               "Work of IDEAL COMPRESSOR"
W_dot_c=W_dot_s_c*eta_c                     "Work of actual compressor"
h[2]=h[1]+(W_dot_c/m_dot)
```

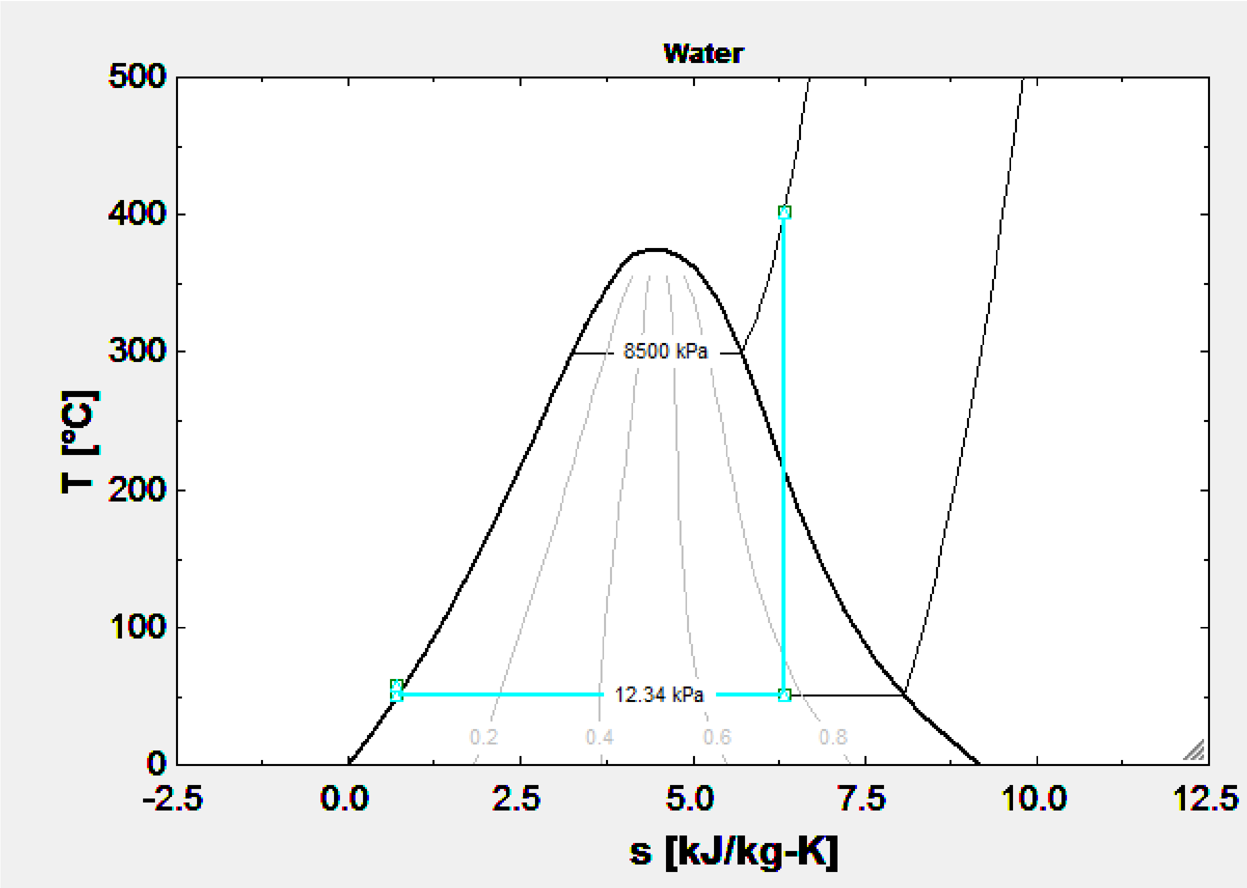
```
T[2]=temperature(R$,h=h[2])
s[2]=entropy(R$,h=h[2],P=P[2])

"State3 Intercooling"
T[3]=T[1]
P[3]=P[2]
h[3]=h[1]
s[3]=entropy(R$,T=T[3],P=P[3])          "assumed P is higher hopefully better entropy?"

"State4 End of Intercooling/Inlet of Reactor"
"T[4]=(T[3]T[5])^.5"
"P[4]=P[3]*(rp^.5)"
P[4]=P[1]*(rp)
s_s[4]=s[3]
h_s[4]=enthalpy(R$,s=s_s[4],P=P[4])
W_dot_s_c2=m_dot*(h_s[4]-h[3])
W_dot_c2= W_dot_s_c2*eta_c
h[4]=h[3]+(W_dot_c2/m_dot)
T[4]=temperature(R$,h=h[4])
s[4]=entropy(R$,T=T[4],P=P[4])
W_in=W_dot_c2+W_dot_c                   "work in from both compressors??"

"State 5 Turbine Inlet"
P[5]=P[4]
 s[5]=entropy(R$,T=T[5],P=P[5])
h[5]=enthalpy(R$,T=T[5])

"State6 Turbine Out"
P[6]=P[5]/rp
s_s[6]=s[5]
h_s[6]=enthalpy(R$,s=s_s[6],P=P[6])
W_dot_s_t=(h[5]-h_s[6])*m_dot
W_dot_act_t=W_dot_s_t*eta_t             "Turbine Work"
h[6]=h[5]-(W_dot_act_t/m_dot)
s[6]=entropy(R$,h=h[6],P=P[6])
 T[6]=temperature(R$,h=h[6])

Q_in=m_dot*(h[5]-h[4])                  "heat in from reactor"
"W_Tout=(h[5]-h[6])*m_dot"              "work from Turbine"
W_net=W_dot_act_t-W_in                  "Net Work"
eta_thermal=W_net/Q_in                  "thermal efficiency"
```

Results

Sort	1 h_i [kJ/kg]	2 P_i [kPa]	3 s_i [kJ/(kg*C)]	4 T_i [C]	5 V_i [m3/kg]	6 $h_{s,i}$ [kJ/kg]	7 $s_{s,i}$ [kJ/(kg*C)]
[1]	293.4	101.3	5.678	19.85	0.83		
[2]	387	315.5	5.629	112.8		406.2	5.678
[3]	293.4	315.5	5.352	19.85			
[4]	387	982.3	5.303	112.8		406.2	5.352
[5]	1130	982.3	6.396	799.9			
[6]	652.9	101.3	6.482	370		599.8	6.396

APPENDIX H

Bottoming Rankine Cycle



//State 1

```
Pw[1]=8500
Tw[1]=T[6]-50
hw[1]=Enthalpy(Water,T=Tw[1],P=Pw[1])
sw[1]=Entropy(Water,T=Tw[1],P=Pw[1])
vw[1]=Volume(Water,T=Tw[1],P=Pw[1])
```

//State 2

```
Tw[2]=50
hw[2]=Enthalpy(Water,s=sw[1], T=Tw[2])
sw[2]=sw[1]
Pw[2]=Pressure(Water,s=sw[1], T=Tw[2])
vw[2]=Volume(Water,T=Tw[2], s=sw[2])
```

//State 3

```
Tw[3]=Tw[2]
hw[3]=Enthalpy(Water,P=Pw[2],x=0)
Pw[3]=Pressure(Water,h=hw[3],x=0)
sw[3]=Entropy(Water,T=Tw[3],x=0)
vw[3]=Volume(Water,T=Tw[3],x=0)
```

```
//State 4

hw[4]=Enthalpy(Water,v=vw[3],P=Pw[1])
Tw[4]=Temperature(Water,v=vw[3],P=Pw[1])
vw[4]=vw[3]
sw[4]=sw[3]
Pw[4]=Pw[1]

//Heat in

q_inw=(hw[1]-hw[4])

//Mass Flow
mw=q_inR/q_inw


//Power Rankine
network=(hw[1]-hw[2])-(hw[4]-hw[3])
powerR=mw*network
eta=network/q_inw

//Total Efficiency
eta_total=(powerR+W_net)/Q_in
```

APPENDIX I

Shell and Tube Heat Exchanger

```
"PRIMARY FUEL HEAT EXCHANGER-VNwadeyi"
"Constants"
q_dot=642738*10^3 [J/s]                          "heat output"
P=982.3 [kPa]                                    "operation pressure"

NT=10 [-]                                        "number of tubes"
MT=8 [-]                                         "number of passes"

R$ = 'air'                                       "material 1"
cp_air= 1014 [J/kg-K]                            "specific heat of air"
m_dot_air= 865 [kg/s]                            "mass flow rate"
mu_air = 230E-07  [N*s/m^2]                      "viscosity of air @ 400 C"
T_Ai=385.8 [K]                                   "inlet temp of air"
T_Ao=T_Ai+(q_dot/(m_dot_air*cp_air))             "outlet"
T_ma= (T_Ai+T_Ao)/2                              "mean temp of air"
k_air= 33.8E-03 [W/m*K]                          "thermal conductivity"

T_Fi= 1273 [K]                                   "inlet fuelsalt temp"
T_Fo=1088 [K]                                    "outlet ' ' "
T_mf=(T_Fi+T_Fo)/2                               "mean temp"
cp_f=2177.14 [J/kg*K]                            "specific heat of fuelsalt"
k_f=0.87 [W/m*K]                                 "thermal conductivity of fuelsalt"
mu_f=0.007 [N*s/m^2]                             "viscosity of fuelsalt"

D_i = 0.035 [m]                                  "inner pipe diameter"
D_o=0.10 [m]                                     "outer"

D_h=D_o-D_i

"Calculations"

m_dot_fuel=q_dot/(cp_f*(T_Fi-T_Fo))              "mass flow rate of fuelsalt"

"Corrected Mean Temp Diff"
Rf=(T_Fi-T_Fo)/(T_Ao-T_Ai)
Pf=(T_Fi-T_Ai)/(T_Ao-T_Ai)
F=0.96                                           "Used 11.11"
MTD= ((T_Fi-T_Ao)-(T_Fo-T_Ai))/ln((T_Fi-T_Ao)/(T_Fo-T_Ai))
    "Mean Temperature Difference"
CMTD=F*MTD
    "Corrected MTD"

mu_a=viscosity(R$,
T=T_ma)
    "real viscosity of air"
Re_a=
(4*m_dot_air)/(pi*D_i*mu_a)
    "Reynolds number of air"
Pr_a=Prandtl(R$,T=T_ma)
    "Prandtl number of air flow"
Nu_a=
0.023*(Re_a^(4/5))*(Pr_a^0.4)
    "Nusslet number of air flow"
h_i=Nu_a*k_air/D_i
    "heat transfer coeff of air"
```

Re_f=4*m_dot_fuel/(pi*(D_o+D_i)*mu_f)
"reynolds number of fuel"
Pr_f=17.513
"prandtl number of fuel salt"
Nu_f=0.023*(Re_f^(4/5))*(Pr_f^0.4)
"Nusslet number of fuel flow"
h_o=Nu_f*k_f/D_h
"heat transfer coeff of fuel"

U=1/(1/h_i +
1/h_o)
"overall heat transfer coeff"

L=q_dot/(U*pi*D_i*NT*CMTD)
"length of tubes"

A=
NT*pi*D_i*L
"area of heat exchanger"
SL=L/MT "

APPENDIX J

Pump Work of ThorCon Breeder Reactor

```
Inlet and Outlet Pump Work of Reactor to Heat Exchanger"

"Constants"
m_dot_fuel=1595 [kg/s]                          "mass flow rate of fuel-HX"
rho_700c = 2010 [kg/m^3]                        "density at 700 C -ORNL-assumed for just
NaFBeF salt"
mass_salt = 0.00889 [kg]                        "molar mass of Salt"
 NT=10
MT=8

H_chimney=3.913 [m]                             "Height of Heat Exchanger- EES HX"
D_h=0.11 [m]                                    "pipe size = shell side of HX"
D_i=0.035 [m]
A=3.14*(D_h^2)/4                                "area of pipe"
eta_pump=0.75 [-]                               "efficiency of pump-assumed"
f=0.0023 [-]                                    "friction factor-calculated for Reynolds # of Fuel
Salt"
g=9.81 [m/s^2]                                  "gravity"

H_core= 5.96 [m]                                "height of core "
r_core=2.75 [m]                                 "radius of core"
T_i= converttemp(C,K,1000[C])                   "inlet temp of HX"
T_o= converttemp(C,K,815[C])                    "outlet temp of HX"

T_meanHX=(T_i+T_o)/2                            "avg temp across HX"

T_reactorin=converttemp(C,K,750[C])             "inlet temp of reactor"
T_meanr= (T_reactorin+T_i)/2                    "average temp of reactor"

"Pressure Drop Across Reactor and Heat Exchanger"

Volume_700C = mass_salt/rho_700c                "Volume used in PV=NRT"
AvgVolume =(Volume_700C/973)*T_meanr            "average volume of salt in core"
rho_core_avg= mass_salt/AvgVolume               "average salt density"

VolumeHX= (Volume_700C/973)*T_meanHX            "average volume of salt in HX"
rho_HX_avg=(mass_salt)/VolumeHX                 "average density in HX"
Delta_P=((rho_core_avg*H_core)-(rho_HX_avg*H_chimney))*g
    "Pressure Drop [Pa]"

"Velocity of Salt Flow"

 Volume_pipe=(Volume_700C/973)*T_i              "volume of salt"
rho_pipe=(mass_salt)/Volume_pipe                "density in Pipe"
Velocity=m_dot_fuel/(2*A*rho_pipe)              "velocity of fuel salt [m/s]"

e_mech=
((Delta_P/rho_700c)+((Velocity^2)/2)+(g*(H_chimney)))*(1/1000)
    "mechanical energy of pump [Kj/kg]"

"Pump Work to HX"

E_dot_mech= m_dot_fuel*(e_mech)                 "mechanical work of pump [kW]"

W_pump_elec = E_dot_mech/eta_pump               "electrical work needed of pump [kW]"
```

"Pump Work to Reactor"

```
Delta_p2pipe= f*(rho_pipe/2)*((Velocity^2)/(r_core+(H_core/2)))*D_h "delta_P in pipe to reactor [Pa]"
e_mech2= ((Delta_p2pipe/rho_700c)+((Velocity^2)/2)+(g*r_core))*(1/1000)
E_dot_mech_pump2=m_dot_fuel*e_mech2
                                                        "kW"
W_pump2_elect
=E_dot_mech_pump2/eta_pump
                                                        "kW"
```

APPENDIX K

Freeze Valve Analysis

Volume Equation Derivation

$$> \ VolumeRate := \frac{dV}{dt} = -\frac{k\cdot(2\cdot\mathrm{pi}\cdot r^3 + V)}{r}\cdot\frac{T}{\mathrm{rho}\cdot L_f}$$

$$\frac{dV}{dt} = -\frac{k\,(2\,\pi\,r^3 + V)\,T}{r\,\rho\,L_f}$$

$$sepvariables := \frac{VolumeRate}{(2\cdot\mathrm{pi}\cdot r^3 + V)}$$

$$\frac{dV}{(2\,\pi\,r^3 + V)\,dt} = -\frac{k\,T}{r\,\rho\,L_f}$$

$$sepv2 := sepvariables\cdot dt$$

$$\frac{dV}{2\,\pi\,r^3 + V} = -\frac{dt\,k\,T}{r\,\rho\,L_f}$$

$$vrate := \left(\frac{lhs(sepv2)}{dV}\right)$$

$$\frac{1}{2\,\pi\,r^3 + V}$$

$$trate := \frac{rhs(sepv2)}{dt}$$

$$-\frac{k\,T}{r\,\rho\,L_f}$$

$$VolFunTime := vrate = trate$$

$$\frac{1}{2\,\pi\,r^3 + V} = -\frac{k\,T}{r\,\rho\,L_f}$$

$$intVolofTime := int(lhs(VolFunTime),\ V)$$

$$\ln(2\,\pi\,r^3 + V)$$

$$intTime := int(rhs(VolFunTime),\ t)$$

$$-\frac{k\,T\,t}{r\,\rho\,L_f}$$

$$Voft := intVolofTime = intTime + C$$

$$\ln(2\,\pi\,r^3 + V) = -\frac{k\,T\,t}{r\,\rho\,L_f} + C$$

$$Vinrh := subs(V = \mathrm{pi}\cdot r^2\cdot h,\ Voft)$$

$$\ln(h\,\pi\,r^2 + 2\,\pi\,r^3) = -\frac{k\,T\,t}{r\,\rho\,L_f} + C$$

$$BoundCond := subs(t = 0,\ h = h_o,\ r = r_o,\ Vinrh)$$

$$\ln(h_o\,\pi\,r_o^{\ 2} + 2\,\pi\,r_o^{\ 3}) = C$$

$$Voft1 := subs(C = lhs(BoundCond),\ Voft)$$

$$\ln\left(2\pi r^3 + V\right) = -\frac{kTt}{r\rho L_f} + \ln\left(h_o \pi r_o^2 + 2\pi r_o^3\right)$$

$$Vt := V(t) = solve(Voft1, V)$$

$$V(t) = e^{\frac{\ln\left(h_o \pi r_o^2 + 2\pi r_o^3\right) r\rho L_f - kTt}{r\rho L_f}} - 2\pi r^3$$

$$Vt2 := simplify(Vt)$$

$$V(t) = e^{\frac{\ln\left(\pi r_o^2 \left(2 r_o + h_o\right)\right) \rho r L_f - kTt}{r\rho L_f}} - 2\pi r^3$$

Iterative Work

ho	1.3	m								
ro	0.05	r(t)		m						
Lf	0.26	J/kg								
k	0.87	J/m*k*s								
T	350	K								
rho	5588	kg/m^3						V(t)		
					1st term (ln(pi*ro^2(2*ro+ho))	2nd Term= kTt/r(t)*rho*Lf	3rd Term= 2*pi*r(t)^3	Value of V(t)		741.7695
t(s)	0	r(t)	0		-4.510262425	0	0	0.010995574	total volume of pipe	12.362825
	1	0.016667	0.05			0.209583723	0.000785398	0.008131152		
	2	0.033333	0.035355			0.419167447	0.00027768	0.006952944		
	3	0.05	0.028868			0.62875117	0.00015115	0.005712321		
	4	0.066667	0.025			0.838334893	9.81748E-05	0.004656641		
	5	0.083333	0.022361			1.047918617	7.02481E-05	0.003785535		
	6	0.1	0.020412			1.25750234	5.34396E-05	0.003073299		
	7	0.116667	0.018898			1.467086064	4.24075E-05	0.002493133		
	8	0.133333	0.017678			1.676669787	3.471E-05	0.002021415		
	9	0.15	0.016667			1.88625351	2.90888E-05	0.001638268		
	10	0.166667	0.015811			2.095837234	2.48365E-05	0.001327259		
	11	0.183333	0.015076			2.305420957	2.15279E-05	0.001074916		
	12	0.2	0.014434			2.51500468	1.88937E-05	0.000870236		
	13	0.216667	0.013868			2.724588404	1.67562E-05	0.000704259		
	14	0.233333	0.013363			2.934172127	1.49933E-05	0.000569693		
	15	0.25	0.01291			3.14375585	1.35193E-05	0.000460616		
	16	0.266667	0.0125			3.353339574	1.22718E-05	0.000372215		
	17	0.283333	0.012127			3.562923297	1.12051E-05	0.000300583		
	18	0.3	0.011785			3.772507021	1.02845E-05	0.000242552		
	19	0.316667	0.011471			3.982090744	9.4833E-06	0.000195547		
	20	0.333333	0.01118			4.191674467	8.78102E-06	0.000157482		
	21	0.35	0.010911			4.401258191	8.16133E-06	0.000126665		
	22	0.366667	0.01066			4.610841914	7.61125E-06	0.000101723		
	23	0.383333	0.010426			4.820425637	7.1203E-06	8.15409E-05		
	24	0.4	0.010206			5.030009361	6.67995E-06	6.52174E-05		
	25	0.416667	0.01			5.239593084	6.28319E-06	5.20199E-05		
	26	0.433333	0.009806			5.449176807	5.9242E-06	4.1355E-05		
	27	0.45	0.009623			5.658760531	5.59815E-06	3.27416E-05		
	28	0.466667	0.009449			5.868344254	5.30094E-06	2.57896E-05		
	29	0.483333	0.009285			6.077927978	5.02913E-06	2.01829E-05		
	30	0.5	0.009129			6.287511701	4.77978E-06	1.56652E-05		
	31	0.516667	0.00898			6.497095424	4.55038E-06	1.20289E-05		
	32	0.533333	0.008839			6.706679148	4.33875E-06	9.10573E-06		
	33	0.55	0.008704			6.916262871	4.14304E-06	6.75939E-06		
	34	0.566667	0.008575			7.125846594	3.96161E-06	4.87941E-06		
	35	0.583333	0.008452			7.335430318	3.79304E-06	3.37633E-06		
	36	0.6	0.008333			7.545014041	3.6361E-06	2.1777E-06		
	37	0.616667	0.00822			7.754597764	3.48969E-06	1.22484E-06		
	38.7695	0.646158	0.00803			8.125456163	3.25353E-06	1.69023E-10	for 1.3 m pipe	total volume
	39	0.65	0.008006			8.173765211	3.22473E-06	-1.24477E-07		
	40	0.666667	0.007906			8.383348935	3.10456E-06	-5.90498E-07		
	41.896	0.698267	0.007725			8.780719674	2.89622E-06	-1.20655E-06	for 2.3 m pipe	total volume
	42	0.7	0.007715			8.802516381	2.88547E-06	-1.23223E-06		

APPENDIX L

Fuel Cycle Enrichment Analysis

```
"Enrichment Calculations"

"This script is used to compute the amount of feed and ore required to fabricate the p = 3520 kg of
enriched Uranium"

x_p =
0.195
        "Product Concentration (19.5% enriched Uranium)"
x_f=0.00711
        "Feed Concentration (natural Uranium)"
x_w =
0.0003
        "Tails Concentration"
P =  3520
[kg]
        "Uranium Required for Startup"
F/P = (X_p-X_w)/(X_f-X_w)
W/P = (X_p-X_f)/(X_f-X_w)
S/P = (2*x_p-1)*ln(x_p/(1-x_p))+((X_p-X_f)/(X_f-X_W))*(2*x_w-1)*ln(x_w/(1-x_w))-((x_p-x_w)/(x_f-
x_w))*(2*x_f-1)*ln(x_f/(1-x_f))
Ratio =
(238*3)/(2*238+8*8)
    "The amount of U238 in U2O8 that is mined"
Mined = F/Ratio
Mined_lb = Mined*convert(kg,lbm)
```

Unit Settings: SI C kPa kJ mass deg

F = 100638	Mined = 76113	$Mined_{lb}$ = 167800
P = 3520 [kg]	Ratio = 1.322	S = 300342
W = 97118	x_f = 0.00711	x_p = 0.195
x_w = 0.0003		

APPENDIX M

Power Distribution of ThorCon Breeder Core

> $a := q'''' = G[f] \cdot N[ff] \cdot \text{sigma}[f] \cdot \text{phi};$

$$a := q'''' = G_f N_{ff} \sigma_f \phi$$

> Power conversion at 280 cm axially and 1cm radially
> Flux at this position:1.67e11 n/s
> G(f)=180 Mev/fission
> N(ff)=?
> sigma(f)=525 barns
>
> N(ff)

> $b := N[ff] = \frac{A[v]}{M[ff]} \cdot r \cdot \text{rho}[fm] \cdot f \cdot i;$

$$b := N_{ff} = \frac{A_v \, r \, \rho_{fm} \, f \, i}{M_{ff}}$$

> A(v)=0.60225e24 molecules/gm mole
> rho(fm)=6.7 g/cc
> r=enrichment percentage 20%
> M(ff)=233.0395 molecular mass
> i=0.53467 fuel atoms per molecule of fuel
> f=?
>
> f=mass fraction of fuel in fuel material

> $c := f = \frac{(r \cdot M[ff] + (1 - r) \cdot M[nf])}{r \cdot M[ff] + (1 - r) \cdot M[nf] + 4 \cdot M[fo]};$

$$c := f = \frac{r M_{ff} + (1 - r) M_{nf}}{r M_{ff} + (1 - r) M_{nf} + 4 M_{fo}}$$

> $d := subs(r = .2, M[ff] = 233.0395, M[nf] = 238.0508, M[fo] = 18.99, c)$

$$d := f = 0.7573229152$$

> Solving for N(ff)

> $e := subs(A[v] = 0.60225e24, r = 0.2, \text{rho}[fm] = 6.7, f = 0.75732, i = 0.53467, M[ff] = 233.0395, b)$

$$e := N_{ff} = 1.402223723 \; 10^{21}$$

> Solving for q"

> $f := subs(G[f] = 180, N[ff] = 1.4e21, \text{sigma}[f] = 525e-24, \text{phi} = 1.67e11, a)$

$$f := q'''' = 2.20941000 \; 10^{13}$$

> Conversion factor
> Converting Mev/s to J/s=Watts

> $g := f \cdot 1.602e-13;$

$$g := 1.602 \; 10^{-13} \; q'''' = 3.539474820$$

> At position of 280 cm in hieght and 1 cm radially, it accounts for 3.539 watts of power.

> To find the total power across our system this same procedure is done for the entire active core region from top to bottom in our system. Our target power level is 642 MW-th power.
> Results:
> System dimensions: Active core hieght: 540 cm (from bottom to top) Actve core(fuelsalt)radius:95 cm
> Power at each hieght(3cm radially):
> 0cm=2456.59 W
> 60cm=6960.98 W
> 120cm=10796.41 W
> 180cm=13190.74 W
> 240cm=13647.32 W
> 300cm=11948.69 W
> 360cm=4006.19 W
> 420cm=650.12 W
> 480cm=88.77 W
> 540cm=3.53 W
>
> To get the entire power across the core we must use the trapezoid rule for the cosine curve of the power distribution.
> Trapezoid Rule:

> $h := \int_{0}^{500} P(z)\,\mathrm{d}z = \frac{\mathrm{delta}(x)}{2} \cdot (2456.59 + 2\cdot 6960.98 + 2\cdot 10796.41 + 2\cdot 13190.74 + 2\cdot 13647.32 + 2\cdot 11948.69 + 2\cdot 4006.19 + 2\cdot 650.12 + 2\cdot 88.77 + 3.53)$

$$h := \int_{0}^{500} P(z)\,\mathrm{d}z = 62519.28000\,\delta(x)$$

> $i := \mathrm{delta}(x) = \frac{(540-0)}{numberofpoints}\Big);$

$$i := \delta(x) = \frac{540}{numberofpoints}$$

> $j := subs(numberofpoints = 10, i)$

$$j := \delta(x) = 54$$

> $k := subs(\mathrm{delta}(x) = 54, h)$

$$k := \int_{0}^{500} P(z)\,\mathrm{d}z = 3.376041120\ 10^{6}$$

> Using the trapezoid rule it was calculated that the summataion for the total active core height to be 3.37e6 Watts for only 1cm radially of the core
> To find the power over the entire core simply do the following:
> Active core radius (fuelsalt):95cm
> Active core diameter(fuelsalt):190cm

> $l := k \cdot 190;$

$$l := 190 \left(\int_{0}^{500} P(z) \, dz \right) = 6.414478128 \; 10^{8}$$

> **Target Power 642 MegaWatts**

> **Caculated Power across core: 6.41e8 Watts or 641 MegaWatts**

APPENDIX N

Email Correspondence with ThorCon Power

Advisor Lars Jorgensen

Initial Design Parameter Correseponedence

From Valerie: Feb 22

Hello Lars,

I'm not sure if you were aware of our UCB counterpart not being able to continue work on the project. When my group last spoke to them they were planning to speak with you to decide on a good time to skype or google chat with you about the chemical composition of the fuel and other design specifications you would like to see us use.

If google chat or skype is not good for you. We can converse via email.

My group would like to discuss our design with you as we saw in your slides that you were opposed to using a brayton Cycle. Our design operates on an open brayton cycle with a bottoming rankine cycle. If you could give some insight or suggestion to this extent that would be helpful.

We will also like for you to expand on the developing country (Indonesia) aspect of ThorCon's 'mision'. I recalled you talking about it on the day we were assigned our project, could you elaborate on Indonesia's power needs, and the advantages and disadvantages of building our plant there.

Thank you and hope to hear from you soon.
Val

Valerie E. Nwadeyi
South Carolina State University Student
Nuclear Engineering Major
Multiple Myeloma Cancer Survivor
(843)-442-3148

From Mr. Jorgensen: Feb 22

'll be traveling tomorrow (to North Carolina) but Thursday morning I'm available waiting for my plane. Cell phone (831) 406-0413, google hangout and Skype are other possibilities.

As far as Brayton versus steam cycle. Steam cycle has a near term advantage of being a very established and competitive industry as we are compatible with coal steam cycles. Brayton is better when the temperatures get higher but for now it is about a wash. (This is also outside my expertise so I'm repeating what I've heard). But for a paper study you could go with either - you don't have the same severe time constraints that we have.

Indonesia has a national goal for 5GW renewable and new power in the next 5 years. Nuclear falls under new power. They also have a goal of 35GW additional power in the next 5 years (likely almost all coal). To reach electricity levels like Europe they will need around 250GWe new power. So there is plenty of growth potential as their economy expands.

The advantages are that they are more open to new nuclear power than the US or EU and they must build out their grid but don't like the choice of coal. The US and EU have sufficient supplies for now. They have a favorable impression of thorium nuclear power and they have a government that wants to get things done. In the US we are more risk averse and are irrationally afraid of radiation. On the negative side, culturally they don't accept financial risks the way the US does. And it is a long ways away.

Their regulator have been working up to be able to regulate nuclear power for a long time but haven't had a power plant to do it on. So this is both a challenge in that there is a lot for them to learn and an advantage. The US regulator is very strongly biased toward LWRs. The rules are written presuming an LWR and it will take a considerable effort to get different rules written.

From Valerie: Feb 22

Also, I saw on the CAN-design picture that the fuelsalt flow rate was 2934 kg/s, is this based on the Thorium decay time to U-233 (27 days)? Or what parameters were used to calculate this?

Thank you again for the help.

Val

From Mr. Jorgensen: Feb 22

The salt flow rate is based on the power level.

Power = (salt heat capacity) * (delta T) * (flow rate).

We are targeting 557 MWth.

Delta T is set by keeping the salt reasonably fluid so can't go much lower than 560C and the

materials for the piping, HX, etc which sets an upper limit around 700C.

Lots of promise that in the future we can push the upper temperature up but we get good enough economics and something we are pretty sure we can build by staying below 704C.

Making deltaT higher will increase thermal transport of materials. Often you have a higher solubility of the metal components (such as chrome) at a higher temperature. The result is dissolving the metal at the hot end of the loop and depositing it at the cold end. Making deltaT higher will accentuate that.

Freeze Valve Analysis Correspondence

From Valerie: Mar 14

Hello Lars,

I am starting to work on the thermodynamic analysis of the freeze valve mechanism. I understand that if the reactor was to over heat that the freeze valve should melt and let the core drain into the decay tank.

I wanted to calculate how long this would take, At what temperatures is the freeze valve expected to melt and what material is it made of?

Is there more to this scenario?

Thank you.

Val

From Mr. Jorgensen: Mar 14

HI Valerie,

Yes. I think you'll find that a single stage freeze valve is not very effective as a safety system. If you have a single stage freeze valve then I think you need to actively cool and/or heat it to get it to behave reliably. So the first simple calculation is to make a single stage valve work as an active component. Perhaps Jack has some details on pipe sizing and volume of fuel salt frozen to calculate how much energy needs to be input to the valve to open it on command.

The more interesting problem is one we haven't designed yet - so you get to be creative :).

Max passive approach (but maybe doesn't work out when we run the numbers)
Imagine a valve high up in the can just outside the reactor (perhaps between the pot and the PHX).
On an overheat situation the top of the pot should heat up more than the bottom (since we no longer have forced circulation) (my guess I haven't done this math yet).

This would then let hot salt flow down and surround the lower valve melting it. Design parameters would be flow rate of the upper salt around the valve trading off upper salt that is hotter since it had less time to cool around the lower valve against transferring more heat into the lower valve) and how far from the top of the pot the upper valve is located (trading off hotter salt against more volume of salt lower down).

More certain less passive approach:
Use a thermocouple attached high on the pot to sense when things are getting too hot. This opens to disable a cooler that is keeping the fuse valve frozen.

Even more certain, even less passive approach:
Use a thermocouple attached high on the pot to sense when things are getting too hot. This closes to enable a battery driven heater to melt the fuse valve.

Other ideas are possible too.

You might write up a few approaches, qualitatively highlighting the likely strengths and weakness of each, then pick one to properly analyze (based in part on what you think will work and partly on what you have the tools to analyze).

Jack and Chris may have ideas as well. Let's trade ideas for a couple of days before setting a direction.

From Dr. Scarlat: **Mar 29**

Valerie,

Thanks for emailing me again. We can meet tomorrow to talk about freeze valve design,
at 1pm or 5.30pm, if that works for you.
For learning COMSOL, I'd suggest touching base with Kazi, Louis, Mohamed, or Jarett, cc-ed on this email.

I'm also cc-ing Francesco, who may be able to point you to literature on freeze valves, and I'm attaching a paper that my group wrote on salt freezing phenomenology. Take a look at the references cited at the end of this paper.

Raluca

Reactor Overheating Analysis

From Valerie: **Apr 3**

Hi Lars,
I am still working on the freeze valve and I have decided to use a thermal couple that signals the active cooling system to stop cooling the valve so that it could melt.

I wanted to know for the ThorCon reactor what would be emergency over heating temperatures. Is it based on the boiling of the fuelsalt or on the graphite?

Thanks for the help

Val

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Nuclear Engineering Major
Multiple Myeloma Cancer Survivor
(843)-442-3148

From Mr. Jorgensen: **Apr 4**

Hi Valerie,
It is based on the creep strength of the vessel. The graphite is good for temperatures in excess of 3000C. The fuel salt boiling is around 1400C. The steel is in trouble by 1100C and has a limited lifetime under stress at 900C.

So part of the question is what temperature should the drain happen at and what kind of margins on this are acceptable.

The top of the reactor is the hottest so I think that is where the thermocouple should go.
Normal operations put the top of the reactor at 704C.
During power ramp transients it may go 20C higher.
So one requirement is that the thermocouple reliably not trigger before 724C.

It takes a few minutes for the reactor to drain. If the thermocouple reliably tripped at 850C then we would protect the primary loop well. Temperature rise rate is around 1C per minute in the worst case when the shutdown rods failed.

So a first draft spec would be to trip nominally within 5 minutes over 800C, reliably not trip at 750C and reliably trip before 850C.

Does that give you the information you need?

Lars

From Valerie: **Apr 6**

Hello,

A quick question from the neutronics stand point, what is the enrichment used in the UF4 in the fuel salt.
Is it initially enriched with U235?
We wanted to make sure that we were operating in MCNP correctly.

Thanks,

Val

From Mr. Jorgensen: **Apr 6**

The reactor can take a variety of fuels. We normally use 19.5% enriched uranium plus lots of thorium. Here is the typical initial fuel for a 12m^3 primary loop used to generate 557 MWth.

Salt Initial composition

Be9	654.833 kg
F19	17163.219 kg
Na23	10579.413 kg
Th232	14050.096 kg
U235	654.676 kg
U238	2652.199 kg

What are you trying to do with the neutronics?
I'm on the road so away from my main computer but I'll try to help if I know what you need

From Valerie: **Apr 6**

Thank you we are actually at 20% enrichment. We were trying to lower our keff/kinf results (with out control rods being inserted) being that we are constructing a breeding reactor. We wanted to make sure that our breeding ratio was near one and that when our fuel leaves the core that it will not be in a critical state. (Hope that makes sense).
So we used ThorCon's fuelsalt precentages and separated them so that it was ThF4-NaFBeF2 in the breeder blanket and then UF4-NaFBeF2 in the main core, but we wanted to makes sure we were operating at normal enrichment percentages.

From Mr. Jorgensen: **Apr +**

Since you earlier asked about graphite I presume you are a thermal reactor. If so, there should be no problem with criticality once you leave the graphite area. I haven't look much at a two fluid reactor for quite a while - I'm not sure how the fuel concentration will change.

I'd guess it depends on your arranging your core geometry so that the right percentage of neutrons leak into the blanket.

As a neutronics computer experiment I think you will learn a lot. In real life there are many other factors to consider
1) keeping the heavy metal fraction near 12% to minimize the melting point of NaBe
2) and keeping the power/unit fissile ratio high to minimize capital costs.
3) keeping the neutron damage to the first wall w/in acceptable bounds
4) be sure that if a major pipe breaks and drains the blanket salt that your reactivity goes down not up (no reflectors outside the blanket).

Anyway, you should learn a lot

My thought exercise is what if the blanket salt drains? I think you will see a dramatic increase in reactivity and likely go serious supercritical.
But since it is a computer run and not a real life test why don't you replace your blanket salt with your cover gas and see what happens?
I think it will be a problem. And I've seen others put a reflector outside the blanket so I don't think it is widely recognized as a problem.
If your simulation shows it to be a real problem and you are so inclined it might even make for a paper

Team LOKI

Neutronics Correspondence

From Samuel: Apr 5

Hi,

My name is Samuel Cole and I am currently doing research with Valerie Nwayedi at the University of Wisconsin on the ThorCon molten salt reactor. I am contacting you in reference to the core design. I have a couple questions and would greatly appreciate if you could answer them to the best of your ability.

1) Since graphite undergoes thermal expansion in the system, is there a gap between the active core and reactor vessel?
If so is it a helium gap, etc.?
2)For the startup flux of your system, was it based from the amount of power you want to produce?
If so could you please give me a ball park number, if applicable.
3) Has there been any cases in which during the graphite thermal expansion, that graphite develops holes in which the fuel salt seeps through?
4) Do you use the empirical burn up model of a PWR light water reactor, or is there a specific burn up model for this type of system?

Thank you

From Mr. Jorgensen: Apr 5

1) Yes there is a gap. It is filled with fuel salt. But the vessel TCE is larger than the graphite TCE so the gap grows as the temperature goes up.
2) We have not chosen a startup neutron source but I expect it makes very little difference. I'd look at ones used for fresh cores in LWRs.
3) The graphite is not used as a salt barrier in our design. We use the Ebasco design (planks are used to form hexagonal logs). See TID-26156.
4) We use Serpent to calculate the fuel evolution. Our fuel lifetime is 8 years.