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Remote Liquid Metal Reactor: A Conceptual Design

Keith Everette Ottinger

University of Tennessee - Knoxville

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Remote Liquid Metal Reactor:

A Conceptual Design

Stephanie A. McKee
Chris Hearn
Jared Hobbs
Kyle Keckler
Keith Ottinger
Kevin Taylor
Bryan Thilking

UTNE Senior Design Class
Spring 2007
Abstract

The Remote Liquid Metal Reactor was designed with the goal of producing a system capable of generating approximately 100 MWe for use in a remote location, while remaining secure against proliferation. It must be easy to transport and install and cost-effective over a thirty year operating life. The proposed design is a lead bismuth-cooled, mixed nitride fuelled core capable of lasting a full thirty years with no refueling and little maintenance. The core, coolant, and steam generators are encapsulated into a single unit, providing for easy transportation and installation, as well as resistance to proliferation. The end result of this effort is that the reactor provides a much needed solution to the problem of providing reliable power in remote areas, while complying with the goals set out by GNEP. This paper outlines the conceptual design of the reactor, the economics and transportation associated with its installation, the safety associated with its operation, and the possible scenarios for decommissioning.
Acknowledgements

Like any endeavor, this work would not have been possible without the assistance of a great number of people: first, Dr. Martin L. Grossbeck, advisor for the UTNE senior design class, and Dr. Ron Pevey, who has acted as both a guest lecturer and frequent advisor; second, the professionals who acted as guest lecturers in the design class: Dr. Ingersoll, Dr. Flannagan, Dr. Juan Carbajo, Dr. Mark DeHart, and Dr. David Cook. Finally, the team would like to thank the designers of the Encapsulated Nuclear Heat Source, for inspiring much of this design.
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Introduction

There are many remote places in the world where electrical power may be required, but where it is difficult to guarantee a constant supply of combustible fuel. In these areas, nuclear reactors provide one possible solution to the task of producing power for a long period of time without needing a constant supply of fuel. The goal of this project is to design a small reactor, which may be transported to a remote location and used to generate approximately 100 MW of electrical power for a long period of time without refueling. The reactor should be economically feasible, so as to be attractive to a small country without its own nuclear power industry, and should be resistant to proliferation, as the remote location in question may make an easy target for a group with interest in securing nuclear fuel. The reactor should also have safety features which would allow it to be successfully licensed by the U.S. Nuclear Regulatory Commission, in order to appeal to a government which may not have a nuclear safety agency of its own. This paper details the design process of a reactor design which meets all of these goals, and gives suggestions for further improvements of the design in the future.

Basic Design

There are a number of components and systems which were integrated, in the following order, to form the complete reactor design: the core, thermal hydraulics, the energy conversion system, shielding, and containment structure. The systems were taken into consideration in this order because it is necessary to perform neutronic calculations in order to size the reactor core prior to in-depth consideration of the other design parameters. Following the selection of LBE as the coolant, it was decided to place the
fuel in pins of 1.0 cm in diameter (with a 0.012 cm He gap and 0.1 cm clad thickness, as shown in Figure 1), spaced at a square pitch of 1.5 cm. These dimensions were chosen in accordance with reference designs of the Encapsulated Nuclear Heat Source, which were used for guidance in choosing initial parameters for the neutronic calculations. As the calculations progressed, it was determined that the rods should be configured in a 70 x 70 lattice (with boron carbide control rods placed at regular intervals), and should contain 8 wt% PuN (100% Pu-239) and 92 wt% UN (natural uranium).

Figure 1: Fuel Pin Slice

The reactor core is located in the bottom half of a large square channel, nested inside a larger cylindrical pool. This vessel also houses a large electromagnetic pump, which is located directly below the core, and the steam generators, which are located around the reactor channel in the top half of the vessel. The steam generators produce superheated steam which is used to turn a turbine that generates electricity. After the steam passes through the turbine it goes to the recuperator, then to a precooler, compressor, intercooler, a second compressor, and then back to the recuperator before it returns to the steam generator to start the cycle over again.
The vessel will be placed below ground level and surrounded by a concrete liner. This configuration was selected for several reasons. First, the presence of the reactor several meters underground and then surrounded by earth means that the reactor is well shielded from the outside world (further discussion to come). Secondly, the placement of the fuel material at the bottom of a long, underground shaft greatly adds to its security, as any person or group with interest in the fuel material would be greatly hindered by the several meters of liquid metal standing between them and the core. Finally, the design of the long cylindrical vessel allows for the vessel to be transported intact, with fuel in place and steam generators inserted through the top of the vessel. This configuration makes the reactor not only more resistant to proliferation, but easy to assemble when the vessel is transported to the reactor site. A sketch of the reactor’s configuration within the ground is shown in Figure 2. A summary table of the reactor’s design parameters is also shown below, in Table 1.

<table>
<thead>
<tr>
<th>Design Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary Coolant Circulation</td>
<td>Natural Circulation</td>
</tr>
<tr>
<td>Average Linear Heat Rate (W/cm)</td>
<td>207</td>
</tr>
<tr>
<td>Core Life (full power years)</td>
<td>30</td>
</tr>
<tr>
<td>Core Height (m)</td>
<td>3</td>
</tr>
<tr>
<td>Core Measurements (m)</td>
<td>1.05 x 1.05</td>
</tr>
<tr>
<td>Fuel Rod Diameter (cm)</td>
<td>1.0</td>
</tr>
<tr>
<td>Clad Thickness</td>
<td>0.1</td>
</tr>
<tr>
<td>Square Pitch</td>
<td>1.5</td>
</tr>
<tr>
<td>Module Height (m)</td>
<td>20</td>
</tr>
<tr>
<td>Module Diameter (m)</td>
<td>1.8</td>
</tr>
<tr>
<td>Weight of Fuelled Module (t)</td>
<td>590 with coolant, 45 without coolant</td>
</tr>
<tr>
<td>Primary Inlet/Outlet Temperature (C)</td>
<td>400 / 639.4</td>
</tr>
</tbody>
</table>
Materials Selection

As is the case with any engineering design, it is crucial to select materials which are well suited to the task at hand. In no situation is this more true than in a nuclear reactor, especially one intended to endure long intervals without refueling or maintenance. In such a taxing environment, it is imperative that the system be cooled by a substance which is resistant to chemical interaction, that the structural materials be...
robust enough to contain the core and coolant while withstanding high radiation levels, and that the fuel material remain intact and resistant to proliferation throughout its operating life.

Coolant

An ideal coolant for a nuclear reactor would have the following properties: low melting point, low probability of corrosion, low neutron absorption cross section, high moderating ratio (for a thermal reactor), high radiation stability, thermal stability, low induced radioactivity, no reaction with the working fluid of the secondary, good heat transfer capabilities, and low pumping power. As no coolant is ideal, it is necessary for the designer to consider the tradeoffs involved in the selection of any particular coolant.

First, the team chose to investigate liquid metals because of their excellent heat transfer properties. While many of the liquid metals are denser than water and gas, and therefore would seem heavier and more difficult to transport, the liquid metals are capable of such effective heat transfer that a reactor utilizing liquid metal coolant requires a significantly lower coolant volume than a water-cooled system. The difference in volume is in fact so great that lead bismuth eutectic-cooled cores end up weighing less than water-cooled cores of the same power rating.

During a continued investigation of liquid metal coolants, the team discovered several properties of lead bismuth eutectic (LBE) which made it more favorable than other liquid metals. First, while sodium, the most popular of the liquid metal coolants, is violently reactive with both air and water, LBE is not vulnerable to any such interaction. Secondly, while sodium melts at a lower temperature than does LBE, sodium also begins to boil at only 883 degrees C. This temperature is not much higher than the desired outlet
temperature of the reactor, and so it would be preferable to select a coolant which does
not boil until a much higher temperature (LBE boils at 910 degrees C). A phase diagram
and summary table of LBE properties are given below. As can be seen, the eutectic
composition of LBE is approximately 56 wt% Bi and 44 wt% Pb.

![Lead Bismuth Phase Diagram](image)

**Figure 3: Lead Bismuth Phase Diagram**

**Table 2: Heat Transfer Properties of Lead Bismuth at 800 K**

<table>
<thead>
<tr>
<th>Property</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Density</td>
<td>10037 kg/m³</td>
</tr>
<tr>
<td>Specific Heat</td>
<td>0.1418 kJ/kg·K</td>
</tr>
<tr>
<td>Thermal Conductivity</td>
<td>0.01686 kW/m·K</td>
</tr>
</tbody>
</table>
**Structural Materials**

While an in-depth investigation of structural materials is somewhat outside the scope of this project, the team thought it prudent to consider the challenges which may be confronted in selecting a material to make up the cladding, vessel walls and other structures. The materials chosen must be able to withstand the high temperatures of normal operation as well as off-normal incidents. These materials should also be resistant to damage caused by the presence of a radiation field, and should not interact with the fuel or coolant. This resistance to corrosion becomes particularly important when the designer considers that the core has a nominal lifetime of thirty years, and that no in-vessel maintenance will be performed during that time. For these reasons, the team chose to consider two steel alloys as possible structural materials; SS316 and MA956, an advanced steel developed for use as an aerospace superalloy. According to Special Metals\(^5\), MA956 is stronger than SS316 in very high temperature LBE environments, because LBE embrittles steels at high temperatures by leaching nickel from them. As can be seen in Table 2 the nickel content of MA956 is very low compared to the nickel content of SS316. After learning this information, it was decided to use MA956 for the cladding and control rod drive mechanisms, because they are exposed to very high temperatures. For the vessel walls and steam generator pipes SS316 was selected because the temperature of the LBE in contact with these components should be low enough to prevent significant nickel leaching and it is much cheaper than MA956.
Table 3: Chemical Makeup of MA956 and SS316

<table>
<thead>
<tr>
<th>Constituent</th>
<th>Weight Percent</th>
<th>MA956</th>
<th>SS316</th>
</tr>
</thead>
<tbody>
<tr>
<td>Iron</td>
<td>Balance</td>
<td>Balance</td>
<td></td>
</tr>
<tr>
<td>Chromium</td>
<td>18.5-21.5</td>
<td>16.0-18.0</td>
<td></td>
</tr>
<tr>
<td>Aluminum</td>
<td>3.75-5.75</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Titanium</td>
<td>0.2-0.6</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Carbon</td>
<td>0.1 max.</td>
<td></td>
<td>0.08 max.</td>
</tr>
<tr>
<td>Yttrium Oxide</td>
<td>0.3-0.7</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Copper</td>
<td>0.15 max.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Manganese</td>
<td>0.30 max.</td>
<td>0.20 max.</td>
<td></td>
</tr>
<tr>
<td>Cobalt</td>
<td>0.3 max.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Nickel</td>
<td>0.50 max.</td>
<td>10.0-14.0</td>
<td></td>
</tr>
<tr>
<td>Phosphorous</td>
<td>0.02 max.</td>
<td>0.045 max.</td>
<td></td>
</tr>
<tr>
<td>Silicon</td>
<td></td>
<td>0.75 max.</td>
<td></td>
</tr>
<tr>
<td>Sulfur</td>
<td></td>
<td>0.03 max.</td>
<td></td>
</tr>
<tr>
<td>Molybdenum</td>
<td></td>
<td>2.0-3.0</td>
<td></td>
</tr>
<tr>
<td>Nitrogen</td>
<td></td>
<td>0.1 max.</td>
<td></td>
</tr>
</tbody>
</table>

Fuel Material

The selection of a material for use as a fuel material presents its own challenges. The team determined that the most effective way to achieve a long lifespan on the fuel would be to use a mixture of plutonium and low-enriched or natural uranium, which would allow for gradual breeding of new fuel. This mixed fuel could be fabricated and used in a number of forms: metallic, oxide, or nitride. The team chose to pursue one of the ceramic fuels, with the hopes that the ceramic fuel form would prove stable over a long irradiation period and would also be more resistant to proliferation than a metallic fuel form. The team chose to utilize the nitride fuel because it provided a number of advantages over the oxide fuels. First, the thermal conductivities of uranium and plutonium nitrides are much higher than those of their oxide counterparts, allowing for much better heat transfer and an overall safer system. Adding to this, nitride fuels have melting temperatures and decomposition temperatures higher than those of oxide fuels.
Finally, nitride fuels have more strongly negative reactivity coefficients than do oxides, further contributing to the safety and stability of the reactor.

**Neutronics**

The SCALE package was used in several forms for performing various calculations throughout the design process. The SAS2H sequence was used to determine the appropriate fuel concentrations, dimensions, and lattice sizes which would yield a core of the appropriate size and lifetime. Once final dimensions were settled upon, the CSAS25 module was used in order to determine reactivity coefficients, and to determine the appropriate size and placement of control rods, which can not be adequately modeled in the SAS2H sequence. Finally, the designers utilized SAS1X to calculate the dose received outside the reactor. This information was then used to determine what type and amount of shielding would be necessary to protect the environment and comply with regulations.

The core, surrounding coolant, reflector, and wall of the inner vessel were modeled in SAS2H, for the purpose of determining the value of the multiplication factor at various points throughout the core’s lifetime. The primary goal in selecting core parameters was to achieve the longest feasible operating time without refueling, while maintaining the lowest possible shipping weight (since additional weight would hinder transportation). The core was assumed to have reached the end of its operational life when $k_{eff}$ dropped below 1.01 (this value was intended to compensate for gaps due to control rods, structural materials, etc.), and the team aimed to stretch the refueling interval to thirty years, while maintaining the smallest possible “footprint” for the inner pool vessel. This small footprint was deemed necessary because the weight of the pool
vessel and core increased dramatically for each small increase in radius of the pool, and because preliminary calculations indicated that it would be simpler to cool a few long fuel rods than a larger number of short ones.

With all of this information in hand, the core was first modeled using the fuel diameters and pitches recommended by the reference cases of the ENHS, and the void gap and cladding thicknesses recommended in class. The core rested inside an 85 cm-radius cylinder of coolant. Original models utilized a beryllium reflector, but it was determined that the reactor could sustain criticality for over thirty years without the aid of the reflector, and so the reflector was removed. Utilizing these dimensions, the smallest core that can remain critical for thirty years is a 70 x 70 array of 300 cm (3 m) long pins. A graph of the fuel pin and assembly values for $k_{eff}$ over time is shown below. A copy of the code used to determine the fuel lifetime may be found in Appendix A. It is important to notice that this code uses a 300 MW thermal power; this value was based on the assumption of 33% efficiency of the energy conversion process. If the efficiency of the energy conversion process turned out to be higher, then so would the final electrical power rating of the plant.
Alternative Pitch

Several additional pitch sizes were used in calculation in order to determine whether a larger or smaller pitch might yield better results. A summary table of these calculations is shown on the next page.
While holding the fuel diameter constant, it was shown that no pitch greater than 1.75 cm can feasibly yield a critical core. A core which had a 1.7 cm pitch had a maximum lifetime of only 4.5 years, far too short for the task at hand. In addition, it was determined that while cores with smaller pitches can last for upwards of thirty years, these cores would be nearly impossible to cool. For example, a fuel rod with gap and cladding was modeled with a total diameter of 1.224 cm, rendering a 1.3 cm pitch only slightly larger than the pin, and complicating cooling.

**Alternative Fuel Diameter and Composition**

Several additional fuel diameters and chemical makeups were also tested to determine whether they might make viable alternatives. A slightly larger fuel pin was tested and would have lasted the thirty year test period, but would have proved difficult to cool and would have also added a great deal to the shipping weight of the reactor, between increased fuel mass and additional coolant needed to keep the reactor at a safe temperature. In addition, the beginning of life core would be supercritical enough that it would prove difficult to control. A smaller diameter fuel was also tested but it only had a lifetime of four years, substantially lower than the design requirements. A summary of these experiments is given in Table 5.

<table>
<thead>
<tr>
<th>Pitch (cm)</th>
<th>Lifetime</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.3</td>
<td>30+ y</td>
</tr>
<tr>
<td>1.5</td>
<td>30+ y</td>
</tr>
<tr>
<td>1.7</td>
<td>4.5 y</td>
</tr>
<tr>
<td>1.9</td>
<td>Not critical</td>
</tr>
</tbody>
</table>

Table 4: Summary of Pitch/Lifetime Calculations for 1.0 cm Fuel Diameter
The effect of using different amounts of UN and PuN in the fuel was also examined. Cores which utilized enriched fuel generally had high beginning of life $k_{\text{eff}}$, but burned out quickly after. For this reason, the decision was made to use natural uranium, with the lowest possible quantities of plutonium, in order to keep the fuel mass down. In the numerous calculations performed, the 92%UN-8%PuN blend in a 70 x 70 pin array appears to provide the longest fuel lifetime with the fewest peripheral problems, i.e. proliferation concerns, extra weight, high costs, and safety issues.

Following the selection of the proper core dimensions, the team investigated the possibility of attempting to flatten the power density of the core. Core designers generally aim to make the power density across the core as even as possible, so that fuel is burned at the same rate everywhere, and the most power is generated for the least amount of fuel. However, the team discovered on plotting the power density of the core that the power density in the active fuel region was nearly flat, with only slight radial decrease due to leakage of neutrons into the coolant. The peak-to-average flux ratio for the core is only 1.38. There would be no advantage changing the makeup of the core because the core already exhibited an unusually flat flux profile. A plot of the flux density is shown on the next page. As mentioned earlier, the active fuel region exhibits a relatively flat shape.

### Table 5: Summary of Fuel Diameter/Lifetime Calculation for 1.5 cm Pitch

<table>
<thead>
<tr>
<th>Diameter (cm)</th>
<th>Fuel Lifetime</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.9</td>
<td>4 y</td>
</tr>
<tr>
<td>1.0</td>
<td>30+ y</td>
</tr>
<tr>
<td>1.1</td>
<td>30+ y</td>
</tr>
</tbody>
</table>
The plot shows a sharp drop-off at the edge of the active fuel region (80 cm radial distance) because of the increased leakage.

Figure 2: Radial Power Density Profile, Final Dimensions

Safety Analysis

In any reactor design, it is crucial to demonstrate that the system can remain safe during an off-normal event. Using neutronic calculations, this can be done in several ways: through determination of the void and coolant temperature coefficients of reactivity and by demonstrating that the system possesses control rods capable of taking the reactor safely subcritical.

As calculated using CSAS25, the relationship between void fraction and $k_{\text{eff}}$ is not linear, though it is usually represented in the literature as such. As can be seen in the accompanying figure, the trend is for decreasing $k_{\text{eff}}$ with increasing void fraction,
meaning that the coefficient must be negative. In order to get a numerical value for the coefficient, a linear fit was performed, with the resulting slope being \(-0.143 \Delta k / \text{void fraction}\). As can be observed from the following graph, this value is less than conservative for low void values, and greatly conservative for higher void fractions (i.e., the slope is always negative, but is sometimes more or less negative than the linear fit would indicate).

\[ y = -0.1427x + 1.068 \]

**Figure 3: Void Fraction vs. K\text{eff} for Final Design**

The relationship between coolant temperature and \(k_{\text{eff}}\) is very nearly linear, with a negative slope. The \(k_{\text{eff}}\) values were again calculated in CSAS25, and are shown in the plot below. The coolant temperature coefficient of reactivity, as calculated using a linear fit, is -2 pcm/degree K.
The fuel temperature coefficient of reactivity, as shown in Figure 8, is also negative, with a value of -1 pcm/degree K, as is to be expected due to thermal expansion of the fuel. While it is unusual for the coolant temperature coefficient of reactivity to be more negative than the fuel temperature coefficient of reactivity, this has been explained in the literature as being due to the properties of the LBE coolant, which reduces the Doppler coefficient due to the harder spectrum it causes.
Following the determination of the coefficients of reactivity, the team needed to demonstrate that the reactor could be safely shut down by the grid of control rods contained in the fuel lattice. The control rods are made of 1.2 cm diameter boron carbide (100% B-10), with a 0.012 cm gap and 0.1 cm thick cladding. They show up as the green boxes in the Figure 9.
Figure 6: 70 x 70 Fuel Lattice with Control Rods

The plot in Figure 10 was generated by simulating a gradual insertion of the control rods. As can be seen, the control rods can be used to take the reactor to a $k_{\text{eff}}$ value of slightly less than 0.9 at the beginning of its life. While regulatory standards only require that the reactor be capable of achieving a $k_{\text{eff}}$ of 0.95 or less at shutdown, there are a few key issues to keep in mind. First, 0.95 is not a limit to which to be designed, but is the absolute limit at which the reactor cannot be licensed. Some margin below 0.95 is not only desirable to safe operation, but also a necessity. Second, it must be recalled that control rods, just like the fuel in a reactor, lose their potency over time. If the system had been designed to a shutdown $k_{\text{eff}}$ of 0.95, or even slightly below, the rods would not be capable of a reliable shutdown very many times before they would require replacement.

Since the control rods are vital to both the safe transportation of the fueled reactor vessel
and to the day to day operation of the plant over a thirty year lifespan, it was decided to aim for a more robust control rod than necessary, in order to ensure safety.

![Control Rod Insertion vs. Eigenvalue](image)

Figure 7: Control Rod Insertion vs. \( k_{\text{eff}} \)

**Thermal Hydraulics**

It was decided to address two major thermal hydraulic issues in this report: steady state full-power cooling and decay heat removal by natural convection. To meet these needs two MATLAB codes were developed. The first of these codes, PumpWork (Appendix C), estimated the work required by the pump to achieve a given mass flux. The second code, HTransfer (Appendix B), calculated the temperatures of the components in the reactor core as a function of height in the core. This program was used to perform two types of calculations. The first type of calculation determined the maximum allowable 2-D horizontal peak to average flux value and the average output...
temperature as functions of the primary coolant total mass flux through the core. The second type of calculation determines the temperature profile for a coolant channel with user specified mass flux and heat flux multiplier.

HTransfer uses the Borishanskii correlation$^4$ to calculate the Nusselt number for the LBE coolant flow. The Borishanskii correlation gives the Nusselt number for metallic fluid flows as

$$Nu = 24.15 \log[-8.12 + 12.76(PI / D) - 3.65(PI / D)^2] + 0.0174[1 - \exp(6 - 6P / D)][Pe - 200]^{0.9},$$

where \( P \) represents pitch, \( D \) represents fuel diameter, and \( Pe \) is the Peclet number, which is the ratio between heat transfer by convection to heat transfer by conduction, and is given by

$$Pe = \frac{V \rho c_p D}{k},$$

where \( V \) is the velocity of the fluid, \( \rho \) is the density of the fluid, \( c_p \) is the specific heat of the fluid, and \( k \) is the thermal conductivity.

The Nusselt number was then used to calculate the heat transfer coefficient and the outer clad temperature. The heat transfer coefficient was defined as

$$h = \frac{Nu \cdot k}{D_H},$$

where \( D_H \) is the hydraulic diameter of the coolant channel, and from this information the outer clad temperature was calculated as

$$T_{c.o} = q_w / h + T_{LBE},$$

where \( q_w \) is the average heat flux from the surface of the fuel and \( T_{LBE} \) is the bulk temperature of the lead bismuth eutectic coolant.
Once the outer clad temperature was known, steady state conduction was used to calculate the inner clad and fuel temperatures, using the equations

\[
T_{c,i} = T_{c,o} + Q \cdot r_{clad}
\]
\[
T_{f,o} = T_{c,i} + Q \cdot r_{He}
\]
\[
T_{f,CL} = T_{f,o} + \frac{Q}{4k_{fuel}} \frac{1}{\pi \cdot dz},
\]

Where \(T_{f,o}\) is the outer fuel temperature, \(T_{f,CL}\) is the fuel centerline temperature, \(dz\) is the incremental change in axial position up the fuel rod, and \(Q\) is the total amount of heat rejected per second (i.e., heat flux multiplied by the surface area of the fuel).

HTransfer continuously updated the coolant and clad properties to improve accuracy of the coolant and clad temperatures. The thermal conductivity of helium, which was used to model the gap, was also continuously updated. The fuel conductivity, however, was not updated because the calculation of thermal conductivity is very complex and is not very important because the fuel temperatures are much too low to cause damage. A summary of the thermal conductivity data used in the calculations is shown below; all values are given as kW/m·K:\n
\[
k_{He} = \frac{0.0476 + 0.362 \times 10^{-3} T_{He} - 0.618 \times 10^{-7} T_{He}^2 + 0.718 \times 10^{-11} T_{He}^3}{1000}
\]

\[
k_{Steel} = 1.43 \times 10^{-5} T_{steel} + 0.007283955
\]

\[
k_{LBE} = 0.00361 + 1.517 \times 10^{-5} T_{LBE}
\]

\[
k_{fuel} = 0.0224
\]
Flux Profiles

The form of the 1-D flux profile for a homogeneous rectangular parallelepiped is:

\[ \cos \left( \frac{\pi x}{L_{ex}} \right) \]

Where: \( x \) = core location between \(-L/2\) and \(L/2\), \( L \) = the core length in the x direction, and \( L_{ex} \) = the extrapolated core length (length over which the neutron flux is non-negligible).

This profile results in a 1-D Peak to Average flux Ratio (PAR) of 1.57 for a bare reactor (ie \( L_{ex} = L \)). This corresponds to a \( L_{ex} \) value of 3.15 m. A horizontal 1-D PAR of approximately 1.4, as yielded during neutronic calculations, yields a horizontal 2-D PAR of 2.1, which is the design limit for this reactor.

Plots of the allowable 2-D horizontal PAR and average coolant outlet temperature both vs mass flux through the core are included as Figure 6 and Figure 7, respectively. The allowable 2-D horizontal PAR was calculated by determining the highest PAR for which the clad temperature did not exceed 1000 degrees C (the MA956 steady state design limit) and the coolant temperature did not exceed 910 degrees C (the boiling point of LBE). These plots show that decreasing the PAR decreases the required mass flux (pumping power) and increases the average coolant outlet temperature (efficiency).
Figure 11. Allowable 2-D Horizontal Peak to Average Flux Ratio vs Core Mass Flux.

Figure 12. Average Coolant Temperature at Core Exit vs Core Mass Flux.
Maximum and Average Temperatures

The mass flux necessary to meet this criteria was determined to be 17000 kg / (m\(^2\) s). This value was then used in two HT transfer calculations of the second type to determine the average and maximum temperatures of the coolant, clad, and fuel. The results of these calculations are shown in Figures 13 and 14 and summarized in Table 6.

![Graph showing temperature variations](image)

**Figure 13.** Average Core Component Temperatures vs Height in Core.
Figure 14. Maximum Core Component Temperatures vs Height in Core.

<table>
<thead>
<tr>
<th>Material</th>
<th>Average Temperature (°C)</th>
<th>Maximum Temperature (°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant</td>
<td>639.4</td>
<td>909.5</td>
</tr>
<tr>
<td>Clad</td>
<td>660.7</td>
<td>943.1</td>
</tr>
<tr>
<td>Fuel</td>
<td>1039</td>
<td>1608</td>
</tr>
</tbody>
</table>

**Pump Work and Natural Circulation**

The PumpWork code gives a very rough estimate of the power that will be required to achieve a specified primary coolant mass flux through the reactor core. The code calculates the pump work by summing the pressure changes resulting from density
gradients, friction losses, and form losses. The equations used to determine the various pressure changes as well as the pump work equation are shown below.

\[ \Delta P_{\text{buoyancy}} = \Delta \rho g \Delta z \]
\[ \Delta P_{\text{form}} = \frac{25G^2}{2 \rho} \]
\[ \Delta P_{\text{friction}} = \frac{fG^2L}{D \rho} \]

where \( \Delta \rho \) is the change in density with height, \( \Delta z \) is the difference in height between the thermal centers, \( G \) is the mass flux of the coolant, and \( f \) is the friction factor, 0.184 multiplied by the square of the Reynolds number.

The losses are only estimates because they require very detailed structural knowledge as well as in-depth calculations, which were not feasible for this project. The pump work required to achieve a mass flux of 17000 kg / (m\(^2\) * s), assuming the pump is 40% efficient, was estimated to be 0.76 MWe. This was calculated using the formula

\[ W = \frac{m \Delta P}{\rho E} \]

where \( \Delta P \) represents the total pressure losses, \( m \) represents the mass flow rate of coolant through the core, and \( E \) represents the pump efficiency.

Natural circulation of the coolant is guaranteed after the initial startup of the reactor because of the location of the core and steam generators. This, however, does not guarantee that reactor core temperatures will stay below design limits if power to the primary coolant pump is lost and the reactor scrams. To prevent the temperatures of the components within the core from increasing above design limits during this accident
scenario the natural circulation mass flux must be greater than or equal to the mass flux required to remove decay heat from the core.

`HTransfer` was used to calculate the required mass flux and average outlet temperature. The calculation of an appropriate heat flux multiplier was also necessary. The heat flux multiplier used was 1.4 (the radial peak to average flux) times 0.06 (the maximum decay power to nominal power ratio) (Todreas). Using these values the mass flux required to prevent the LBE from boiling was determined to be $1020 \text{ kg / (m}^2 \cdot \text{s})$. With this mass flux the average coolant temperature of the LBE at the core exit is 637°C. These values were plugged into `PumpWork` to see if work was required from the pump to achieve the necessary mass flux. The pump work was calculated to be greater than zero meaning the coolant could lose energy in the pump and still flow at the required mass flux.

**Secondary System**

Because the reactor runs at temperatures low enough to make a Brayton or Stirling cycle unattractive, it was decided that a Rankine cycle would be utilized to convert the reactor's heat into useful electrical energy, and that the cycle would use steam as a working fluid. The process begins with the transfer of heat from the primary coolant to the coolant contained by the outer pool. This coolant then sheds heat to the four steam generators submerged inside the edge of the outer vessel. The steam produced in the steam generators is then passed on to a steam turbine, which powers an electrical generator used to produce electricity. From the turbine, the steam passes into a recuperator, and then on to a series of coolers and compressors which allow the steam to shed as much energy as possible before beginning the circuit through the reactor once.
again. These coolers and compressors are configured to improve the overall efficiency of the system by reducing the inlet temperature of the steam generator as much as possible. A schematic of this system is shown below. At the current time, the form of the heat sinks has not been determined, as the form would likely rely on the geographic features of the area in which the system is constructed. If the reactor were to be built near a large body of water, then this body of water would likely provide an adequate heat sink for the system. It is far more likely, however, that the system would be installed far from a source of water, and in this case, the design would employ one or more cooling towers in order to supply water for use in the precooler and intercooler shown in the schematic.
In most areas of the design of the secondary system, so little information was available that it is impossible to completely define the system. Reliable information on the operating limits and efficiencies of turbines, compressors, and coolers was often impossible to obtain from suppliers, and so a great deal of work remains to be done in order to fully establish the design parameters of the secondary system. For this reason, it was assumed that the system was capable of attaining a minimum of 33% efficiency. While it is known that many steam cycles are capable of attaining efficiencies of up to
40%, it was also realized that each increase in efficiency often requires a vast amount of additional equipment. Since one of the primary goals of this design was to make the transport and setup of the system as easy as possible, it was decided that 33% efficiency would be sufficient for the task at hand.

Although few specifics about the secondary system could be determined, the information yielded from thermal hydraulics calculations allowed the following parameters to be set for the steam generators:

<table>
<thead>
<tr>
<th>Table 7: Steam Generator Specifications</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parameter</td>
</tr>
<tr>
<td>Inlet Temperature</td>
</tr>
<tr>
<td>Outlet Temperature</td>
</tr>
<tr>
<td>Mass Flow Rate</td>
</tr>
<tr>
<td>Steam Pressure</td>
</tr>
</tbody>
</table>

In order to convert the steam's energy into electricity, the team selected a turbine from the “Reheat-Combined Cycle A” Series available from General Electric\(^{11}\). The turbine was chosen for a number of key features, including small size, robust construction, high efficiency (exact numbers were not given, but it is known that the reheat capability adds efficiency), and rapid startup capability. This rapid startup capability is important given that the design problem mandates simple setup and installation of the reactor and its subsystems. The turbine also has a power rating perfectly suited to this reactor, generating between 80 and 150 MWe. A discussion with GE personnel also indicated that the turbine's other parameters make it capable of sustaining the inlet temperature, pressure and mass flow rate which correspond with the
outlet of the steam generator described above. A photograph of the turbine is shown below.

![Figure 9: Reheat Combined Cycle A Turbine](image)

After performing work on the turbine in order to produce electricity, the steam moves on to be condensed and cooled in order to start the entire cycle again. It was intended to purchase a commercially available recuperator, precooler, intercooler and compressors to use in cooling the working fluid; unfortunately, very little specific information on these systems was available from GE. After shedding some heat to the working fluid in the recuperator, the fluid passes through a precooler, a compressor, an intercooler, a second compressor, and then the recuperator again, before it starts the cycle over again. This step of passing through the recuperator allows the fluid which is about to be pumped to the steam generator to be slightly preheated while cooling the fluid which has just exited the turbine, and allows for increased efficiency of the secondary system.

While passing through the precooler and intercooler, the fluid would be cooled by water available on site. This water would be present either in the form of a local body of water, or as a loop of water intermediately heated by the hot water entering the
precooler and intercooler, and then cooled by descending through a cooling tower. The latter case would be more likely since it is unknown whether the site of the facility will enable access to a body of water. A picture of the proposed cooling tower is shown below\textsuperscript{12}.

![Cooling Tower Overhead View](image)

**Figure 10: Cooling Tower Overhead View**

**Passive Safety Features**

When designing a reactor that requires limited operator control and which must also possess a large factor of safety, it is critical that the design incorporate passive safety systems. Passive nuclear safety describes a safety feature of a nuclear reactor that requires no operator action and little or no electronic feedback in order to ensure safety in the event of a reactor transient. Passive safety features pertaining to this reactor include
negative temperature feedback coefficient of reactivity, negative void coefficient of reactivity, natural convection cooling and methods for reactor scram.

Void coefficients of reactivity represent how the reactor responds to any void formation in the liquid coolant. A negative number signifies that reactivity will decrease as voids are formed in the primary coolant system. While it is unlikely that voiding would be likely to occur in the reactor due to the very high boiling point of the LBE, it is still crucial to have a negative void coefficient of reactivity, in the even that an unexpected loss of coolant occurs. An example of a positive void coefficient and its detrimental effects were evident during the Chernobyl accident, as a positive void coefficient was one major contribution to the instability of the reactor.

Equally important passive safety features is negative fuel and coolant temperature coefficients of reactivity. These are the measures of how a reactor responds to an increase in either fuel or coolant temperature. When this value is negative, reactivity decreases as temperature increases. This would prove very important in the event that either the fuel or coolant temperature increased suddenly due to insufficient removal of heat. In the event that either the fuel or coolant temperature increased, the resulting negative feedback would cause the reactivity to decrease, and the temperature to decrease.

While the reactor would be able to sustain small changes in temperature without significant safety issues, it is also important to ensure that a sufficient flow of coolant is available under all circumstances. For this reason, it was decided to cool the reactor through natural convection. As discussed in the heat transfer section, the reactor can be cooled by natural convection in all but shutdown conditions. In the event of a shutdown, circulation of the coolant would be maintained by an electromagnetic pump located...
beneath the active core. This ability to cool the reactor without the presence of a pump is important because pumps can occasionally fail, and so it is considered greatly advantageous to be able to cool the core reliably without the aid of a pump.

In the event of a reactor transient or the loss of onsite or offsite power it is critical that a reactor scram is immediately initiated. A reactor scram involves the immediate insertion of all reactor control rods. The intent of this reactor scram would be to immediately stop the fission process thus reducing the amount of heat being produced. The ultimate purpose of this would be to prevent damage to the reactor core. The control rod drive mechanisms in the reactor resemble that of a pressurized water reactor, in that the control rods enter the reactor core from the top of the core. This ensures that even in the event that reactor power is lost the electronic control rod drive mechanisms fail in a safe mode where the control rods are allowed to fall into the reactor core via gravity. This reliance on gravity without the need of operator or mechanical input makes this another passive safety system of the reactor.

**Shielding**

When dealing with fissionable materials, such as those contained inside a nuclear reactor, shielding and safety are major design features. These design features must be implemented properly to ensure the safety of the public and the environment. Although our reactor is designed for deployment in remote regions of the world, shielding and safety must be addressed properly in order to meet current NRC regulations. In addition to the importance of shielding from the biological effects of radioactivity, shielding materials often provide addition protection from outside intrusion. Even though shielding is very important, it is easily achieved by inexpensive and readily available resources.
such as concrete, which serves as a more than sufficient shield from fast neutrons and other forms of radioactivity.

In order to determine the proper amount of shielding required for the design ORNL’s nuclear criticality safety program SCALE was used. More specifically the SCALE sequence SAS1X was used. SAS1X is a combined criticality and shielding calculation that takes the leakage spectrum from an XSDRNPM one-dimensional criticality calculation and uses it as a boundary source in subsequent one-dimensional shielding calculations. Since the reactor must remain safely shielded during both normal operation and transportation prior to installation, it was decided to model the reactor both with and without lead bismuth coolant. Since the most critical geometry is a sphere, the reactor core was modeled as a sphere which consisted of all materials contained within the core. These materials include the fuel, cladding, and the coolant (for the normal operation case). From this the team was able to surround the model with the appropriate materials to shield the surrounding medium from fast neutrons and other sources of radioactivity. In this case borated concrete serves as a sufficient shield from such sources of radioactivity. In order to meet NRC regulations it was determined that 95 cm of concrete and an additional 5 cm of steel prove sufficient to achieve less than the regulation-stated maximum dose of .5 (millirem/hr): the dose during transportation, when the vessel has no coolant, was calculated as 0.46 mrem/hr, and the dose during normal operation was determined to be 0.21 mrem/hr. During transportation, this shield will take the form of a removable cask, and during normal operation, the shield will take the form of the lining for a cylindrical shaft in which the reactor will rest.
**Containment**

The main purposes of a containment building are to reduce radioactive releases to the environment, to prevent or limit potential damage to the system (whether by terrorism, natural disaster or other means), and to mitigate accident conditions by preventing vital equipment from exceeding design and safety limits. In addition, the structure should protect vital equipment from internal and external events, while protecting onsite workers from radiation and providing physical protection (i.e., security) for vital equipment. While none of these six functions is exclusively a containment building system function, the first three may be viewed as mitigative functions, while the last three may be viewed as preventive functions. The design will incorporate an industrial structure to protect vital equipment from potential environmental hazards will add a layer of security to the buried reactor. With the reactor underground, adequate shielding is provided for onsite workers, and release of radiation to the environment is highly unlikely.

The presence of the reactor in an isolated environment in another part of the world also raises concerns that an individual or group may attempt to tamper with the reactor. Burial of the reactor several meters underground therefore protects the reactor vessel itself from intrusion, and the containment structure above ground prevents tampering with other systems and equipment. The proposed industrial structure is constructed of a concrete and steel bar reinforced mixture to ensure protection of reactor vitals. A diagram of the proposed containment structure is shown below.
As the reactor in question must be designed to be installed in a remote location, possible in another country, transportation poses a particularly important problem. While a reactor of its size and its support equipment cannot possibly be transported onboard a single vehicle, the nature of this design is such that it can be transported as individual components on several trucks. The ENHS, on which this design was loosely based, was intended to be transported intact, complete with frozen coolant in place. Since this reactor weighs in at a total of 590 metric tons when filled, the reactor vessel would be too heavy to transport when filled with coolant, as the heaviest tractor trailers can haul only 400 metric tons \(^{14}\). The vessel would therefore need to be transported empty, without coolant, and with all control rods inserted in order to maintain criticality safety in transit (the

---

Figure 18: Proposed Containment Structure\(^{10}\)

**Transportation**

As the reactor in question must be designed to be installed in a remote location, possible in another country, transportation poses a particularly important problem. While a reactor of its size and its support equipment cannot possibly be transported onboard a single vehicle, the nature of this design is such that it can be transported as individual components on several trucks. The ENHS, on which this design was loosely based, was intended to be transported intact, complete with frozen coolant in place. Since this reactor weighs in at a total of 590 metric tons when filled, the reactor vessel would be too heavy to transport when filled with coolant, as the heaviest tractor trailers can haul only 400 metric tons \(^{14}\). The vessel would therefore need to be transported empty, without coolant, and with all control rods inserted in order to maintain criticality safety in transit (the

---

37
absence of coolant actually aids in-transit safety because of the negative void coefficient of reactivity). A test case was run in SCALE to ensure that the vessel could be safely transported, yielding a $k_{\text{eff}}$ of: 0.7531, well below the generally accepted criticality of safety limit of 0.95.

The coolant would then be transported separately, within casks so that it could be easily melted and injected into the reactor for startup. All other materials crucial to the design would also be easily transportable on trucks, including the secondary pool, steam generators, piping, turbine, containment structure and concrete used to construct a lining for the shaft into which the reactor vessel would be inserted. The most difficult part of the transportation operation would remain the movement of the twenty meter long fueled reactor vessel, which would easily fit onto a modified double tractor trailer (22.6 m long). While some concerns have been raised about the safety and legality of moving such large equipment on highways, it should be noted that the weight of the reactor and its associated shielding (approximately 91 metric tons of concrete and steel) is well below the maximum weight limit for truck transport in the U.S. and most European countries.

Economics

The economics of constructing this reactor, while very important to making the design likely to be constructed, are nearly impossible to discuss accurately in such an early design. At this point, the best that can be done is to estimate the costs of materials based on the cost of construction of a larger plant. One source estimated the startup cost of a new nuclear power plant at $2000 per kilowatt. This would lead to an estimated $200 million in startup costs for this design. While this sounds like an immense cost for such a small system, when averaged over a thirty year operating life (assuming no
additional fuel costs due to the thirty year lifetime of the existing fuel), the resulting cost of electricity is 0.76 cents/kWh. This value does seem inordinately small, and would in all likelihood prove lower than the actual cost of electricity produced by the system, as this system is different from a large power station in several respects. First, most large scale power plants do not require transportation cross-country and/or overseas as this system was intended to be transported. Second, the estimates given for construction cost per kilowatt hour do not take into account the economy of scale which accompanies large-scale power production. A small plant would likely experience higher costs per kilowatt than this estimate implies. Finally, this figure only considers construction costs, and not the costs associated with day to day operation, which include security, maintenance, and employee salaries. Nevertheless, the extremely low estimated cost per kilowatt-hour shows that this design is likely to be economically viable.

**Shutdown**

A crucial part in any reactor design is considering what course of action must be taken when the reactor has reached the end of its operational life and is shut down. For this reactor design there are two possible scenarios to consider for shutdown. In each case, the reactor must first shed its decay heat, which will require pumping of the coolant in order to keep the reactor cool, as thermal hydraulic analysis indicated that natural convection would be sufficient means for cooling in every case but during decay heat removal. Only after the fuel has cooled to the melting temperature of the LBE can all pumping be ceased. After this, one scenario involves the use of a drain tank to remove molten lead-bismuth, after the fuel has been cooled to approximately the melting temperature of the LBE. The LBE drain tank could be an insulated steel vessel placed
underground in a location where the highest point of the tank remains below the lowest point of the reactor vessel, or could be placed above ground and employ some kind of vacuum pump to remove coolant. Such systems have been implemented at Los Alamos National Lab, where a heated drain tank was used to hold LBE for the lab’s spallation neutron facility. After removal of the coolant, the encapsulated nature of the reactor means that the entire pool vessel may be pulled from the ground and taken to another site for processing and/or permanent storage of the waste, and that another capsule may be installed in its place to continue generating power. The removal of the coolant may be problematic due to the presence of polonium and other activation products within the LBE, and so another option for the shutdown of the reactor should be considered.

The other possibility is to remove the steam generators and allow the coolant to solidify and entomb the core. This process will allow for adequate shielding as long as the core is stored in place. Because the LBE would render the system too heavy to haul from the site by most means, the best course of action would be to store the spent core in place either temporarily or permanently. Either would probably suffice since the design problem states that the reactor facility would be constructed in a remote location.
Figure 11: Diagram of Potential Underground Coolant Tank

GNEP Guidelines

The Global Nuclear Energy Partnership's (GNEP) plans to begin recycling spent nuclear fuel from commercial reactors in order to reduce the volume of hazardous material. One process incorporates use of fast breeder reactors which would consume or destroy these transuranics produced in commercial reactors. This would reduce the
volume of material needing disposal in Yucca Mountain, which would effectively double the capacity of the geologic repository. These advanced fast reactors will incorporate safety and operational design features from the beginning to protect public and worker health while reducing greenhouse gas emissions during electricity generation. A key objective of the advanced fast reactor program is to obtain design certification from the U.S. Nuclear Regulatory Commission for a standard plant. The reactor has been designed to comply with all areas vital to becoming certified by the Nuclear Regulatory Commission.

The Global Nuclear Energy Partnership (GNEP) calls for an expanded program to design, build, and export nuclear reactors that are cost effective and well suited to conditions in developing nations. This reactor design specifically addresses this market for an essential small electric reactor for developing nations and small-grid markets without increasing proliferation concerns. This design could meet the rising power demands associated with economic growth and urbanization, while avoiding the use of fossil fuels that would otherwise be burned in power plants. The safety features inherent in the proposed reactor ensure reliability, and remote and unattended monitoring technologies, while maintaining advanced containment, smart safeguards, automated control systems and monitoring sensors.

The USA aims to make small integrated power reactors available to developing countries. GNEP literature suggests small reactors could come complete with fuel to last the lifetime of the reactor, like the lead-cooled, transuranic-fuelled ‘portable’ STAR series10. The reactor design fulfills these needs and performs multiple roles to expand the nuclear industry. The only potential GNEP specification problem with our design is the
use of pure plutonium products, but due to the design of the reactor vessel, access to the core is extremely challenging and improbable. In addition, the plutonium nitride makes up only eight percent of the total fuel mass, with the rest taken up by natural-enrichment uranium nitride. This, combined with the fact that it would be difficult to free the plutonium from a ceramic state, would make the fuel of little interest to any unassociated party. Once loaded with coolant it becomes virtually impossible to extract fuel from the vessel without a tedious coolant extraction process, which would require radiation shielding. After shutdown the coolant entombs the core in a blanket of solid lead bismuth, making it even more difficult to access the fuel.

**Future Work**

As is true of any preliminary design, there are a great number of areas still to be explored. First, while researchers are hopeful about many of the materials recommended for use in this design, several are unproven for use over a thirty year irradiation period. Future researchers will want to consider further the damage to these materials (particularly the nitride fuel) over a long lifetime and may recommend alternative materials or treatments that would better serve such a design.

Another area of vast importance which has not yet been explored involves the secondary system. At the time of the release of this document, the team could not gather enough information to make a detailed design of the secondary system possible. It is hoped that in the future more information on commercially available solutions will be found, or that some team in the future would be able to design a system specifically to meet the needs of this reactor. In addition, it is likely that with some additional work, the Rankine cycle used in the secondary might be made to operate with higher efficiency.
through the use of reheating or other techniques, though any improvement of the thermodynamics of the system may result in additional capital cost, or weight to be transported, in the form of additional equipment. As for now, the system is about as economically feasible as is likely to be possible.
References


8. Litfin, K. and Stieglitz, R. *Sub-Channel Analysis of Heavy Liquid Metal Cooled...*


Appendix A: SCALE Input Decks
SAS2H Final Model
=csas2h parm='skipshipdat
'****************************************************
70 x 70 LMFBR
'****************************************************
LMFBR
44groupndf5 Latticecell
un 1 0.92 1800 92235 0.7 92258 99.3 end
pun 1 0.08 1800 end
wtptma956 2 7.85 12
  24000 19
  13027 4
  22000 0.3
  6000 0.05
  39000 0.3
  29000 0.1
  25055 0.2
  27059 0.25
  28000 0.4
  15031 0.01
  26000 75.39
1 1000 end
pb 3 0.44 900 end
bi 3 0.56 900 end
be 4 1.0 900 end
end comp
squarepitch 1.5 1.0 1 3 1.224 2 1.024 0 end
NPIN/ASSM=4900 FUELNGTH=300 NCYCLES=1 NLIB/CYC=10
  PRINTLEVEL=4 INLEVEL=3 NUMZTOTAL=4 end
3 0.001 500 80 3 85 4 90
BON end
NIT end
XSD
Weighted cross sections
l4= -1 3 0 9
x5=.0001 .00001 1.0 0.0 1.42 300 end
POWER=300. BURN=10950 DOWN=15 end
end

CSAS25 Base Case, with Control Rods
'Input generated by GeeWiz SCALE 5.1 Compiled on November 9, 2006
=csas25 parm=(nitawl)
lmfbr general case
44groupndf5
read composition
wtptma956 2 7.85 12
24000 19
13027 4
22000 0.3
6000 0.05
39000 0.3
29000 0.1
25055 0.2
27059 0.25
28000 0.4
15031 0.01
26000 75.39

1 1000 end
pun 2 0.08 900 end
un 2 0.92 900 end
wtptbpi 3 10 2
82000 44
83000 56
1 1000 end
be 4 1 1000 end
b4c 5 1 900

5010 100 end
end composition
read parameter
htm=yes
end parameter
read geometry
unit 1
com='fuel rod and channel'
zcylinder 2 1 0.5 300 0
zcylinder 0 1 0.512 300 0
zcylinder 1 1 0.612 300.1 -0.1
cuboid 3 1 0.75 -0.75 0.75 -0.75 300.1 -0.1
global unit 2
com='fuel, coolant and reflector'
array 1 -38 -38 10
zcylinder 3 1 150 400 -20
zcylinder 4 1 155 400 -30
cuboid 0 1 170 -170 170 -170 410 -40
unit 3
com='control rod'
zcylinder 5 1 0.6 300 280
zcylinder 0 1 0.612 300 0
zcylinder 1 1 0.712 300.1 -0.1
cuboid 3 1 0.75 -0.75 0.75 -0.75 300.1 -0.1
end geometry
read array
ara=1 nux=70 nuy=70 nuz=1

```
com=""
fill
```
```
```
```
```
```
end fill
end array
read bnds
+xb=vacuum
-xb=vacuum
+yb=vacuum
-yb=vacuum
+zb=vacuum
-zb=vacuum
end bnds
end data
end
SASIX Shielding Input

#sas1x parm='size=900000'
spherical reactor with concrete and stainless steel shielding
27n-18couple multiregion

'multiregion must be specified to run combined criticality/shielding problem.

un  1  0.32108  1258 92235 .7 92238 99.3 end
pun 1  0.02792  1258 end
pb  1  0.20988  938  end
pb  1  0.26712  938 end
ss316 1  0.157  1073 end

reg-concrete 2 1.0 end
boron 2 1.0 end
activities 2 0 1.e-24 end

ss316 3 1.0 end
activities 2 0 1.e-24 end

dendcomp

's the criticality calculation input
spherical vacuum end
1 75
end zone

' isn=16 is specified to match the angular quadrature in the shielding calc.
more data isn=16 end more data
end

last
reactor shielding

's the shielding calculation input

spherical
'first mixture must be void of 1 interval with outer dimension that matches
'outer dimension of shielding calculation.
'flags indicate boundary source will be input from xsdrnpm criticality calc.

0 75 1 1 0 0 0
2 170 170 0
3 175 175 0
end zone
read xsdose
end
Appendix B: HTransfer Code
clear
clc

fluxstep = .01;
dz = .01;
dG = 500;  
dG = 500;  
Pitch = .015;  
Pitch = .015;  
FuelOD = .01;  
FuelOD = .01;  
GapOD = .01024;  
GapOD = .01024;  
CladOD = .01224;  
CladOD = .01224;  
FuelHeight = 3;  
FuelHeight = 3;  
T_LB(1) = 400;  
T_LB(1) = 400;  
AxS = Pitch*2 - pi*(CladOD/2)^2;  
AxS = Pitch^2 - pi*(CladOD/2)^2;  
Pw = pi * CladOD;  
Pw = pi * CladOD;  
Dh = 4*Axs/Pw;  
Dh = 4*Axs/Pw;  
Power_nom =300.;  
Power_nom =300.;  
G1 = 17000;  
G1 = 17000;  
z(1) = 0;  
z(1) = 0;  
PD = Pitch / CladOD;

% Square array with s pins per side, m total pins, c control rods, 
% and n fuel rods
s = 70.0;  
s = 70.0;  
m = s^2;  
m = s^2;  
c = 64;  
c = 64;  
n = m - c;  
n = m - c;

AvgP = Power_nom*10^3/n;  
AvgP = Power_nom*10^3/n;  
AvgPl = AvgP / FuelHeight;  
AvgPl = AvgP / FuelHeight;  
q_w0 = AvgPl/Pw;  
q_w0 = AvgPl/Pw;  
L = FuelHeight+.15;  
L = FuelHeight+.15;  
L2 = FuelHeight/ 2;  
L2 = FuelHeight/ 2;

for count = 1:(FuelHeight/dz)
qflux(count) = sin(pi*(-L2 + count*dz)/L) -sin(pi*(-L2 + (count -
 1)*dz)/L);
end

qflux = (qflux / mean(qflux)) * q_w0;
VerticalPeakToAverage = max(qflux)/mean(qflux)

for abc = 0:0
G(abc+1) = G1 + abc*dG;
mdot = G(abc+1) * AxS;

ii = 2;
Tc_i(1) = 1000;
jj = 1;
T_SS = T_LB(1);
T_He = T_LB(1);
T_LB(2) = T_LB(1);
% while (Tc_i(ii-1) <= 1000 & T_LB(ii) < 910)

    qflux2 = qflux * jj;
    jj = jj + fluxstep;
    ii = 1;
    kk = 2;

    while (z(ii) < (FuelHeight - dz) & Tc_i(kk-1) <= 1000 & T_LB(kk) < 910)

        T = T_LB(ii) + 273.15;

        % Thermophysical properties of PbBi for Temperature in Kelvin
        rho_LB = 11096 - 1.3236*T;
        m^3
        cp_LB = .159 - 2.72E-5*T + 7.12E-9*T^2;
        % kJ / kg m^3
        * Y:
        k_LB = 3.61E-3 + 1.517E-5*T +1.741E-9*T^2;
        % kW / m * K
        * K:

        % Thermal conductivity of Fuel (UN92%,PuN8%, He(gap), and
        clad(MA 956)
        % for temperature in Kelvin
        k_fuel = .0224;
        % kW / m * K
        k_He =(.0476+.362E-3*T_He-.618E-7*T_He^2+.718E-11*T_He^3)/1000;% kW / m * K
        k_SS = 1.430000E-05*T_SS + 7.283955E-03;
        % kW / m * K

        % Calculations
        dT = dz * Pw * qflux2(ii) / (cp_LB * mdot);
        % C
        Q = qflux2(ii) * Pw * dz;
        % kW

        v = G(abc+1) / rho_LB;
        % m

        Pe = rho_LB * cp_LB * v * Dh / k_LB;

        if (Pe > 200)
            Nu = 24.15*1og10(-8.12 + 12.76*PD-3.65*PD^2)+.0174*(1-
            exp(6-6*PD))*(Pe-200)^.9;
        else
            Nu = 24.15*log10(-8.12 + 12.76*PD-3.65*PD^2);
        end

        h = Nu * k_LB / Dh;
        % kW / m^2 K
        Tc_0(ii) = qflux2(ii) / h + (T_LB(ii));
    end

    Rclad = log(CladOD/GapOD)/(2*pi*k_SS*dz);
% Tc_i(ii) = Tc_o(ii) + Q * Rclad;
RHe = log(GapOD/FuelOD)/(2*pi*k_He*dz);
Tf_o(ii) = Tc_i(ii) + Q * RHe;
Tf_clad(ii) = Tf_o(ii) + (Q/(dz*pi))/(4*k_fuel);

% Approximate average clad and gap temperatures
T_SS = (Tc_i(ii) + Tc_o(ii))/2 + 273.15;
T_He = (Tc_i(ii) + Tf_o(ii))/2 + 273.15;

if (ii == 1)
    T_SS2 = T_SS - 1;
    T_He2 = T_He - 1;
    while (abs(T_SS - T_SS2) > .1 | abs(T_He - T_He2) > .1)
        k_He = (.0476+.362E-3*T_He-.618E-7*T_He^2+.718E-11*T_He^3)/1000; % kW / m * K
        k_SS = (9.0109 + 1.5298E-2*T_SS) / 1000;
        Rclad = log(CladOD/GapOD)/(2*pi*k_SS*dz);
        Tc_i(ii) = Tc_o(ii) + Q * Rclad;
RHe = log(GapOD/FuelOD)/(2*pi*k_He*dz);
Tf_o(ii) = Tc_i(ii) + Q * RHe;
Tf_clad(ii) = Tf_o(ii) + (Q/(dz*pi))/(4*k_fuel);
T_SS2 = T_SS;
T_He2 = T_He;
T_SS = (Tc_i(ii) + Tc_o(ii))/2 + 273.15;
T_He = (Tc_i(ii) + Tf_o(ii))/2 + 273.15;
end
end

% Update variables
z(ii+1) = z(ii) + dz;
T_LB(ii+1) = T_LB(ii) + dT;

ii = ii + 1;
kk = ii;
end

if (jj < (1+fluxstep*1.5) & (ii > length(qflux)))
    T_out(abc+1) = T_LB(ii-1);
elseif (jj < (1+fluxstep*1.5))
    T_out(abc+1) = 1000;
end

end
p2a(abc+1) = jj - fluxstep;
end

T_LB = T_LB(1:length(T_LB)-1);
z = z(1:length(z)-1);

figure(1)
c1f
plot(z,[T_LB' Tc_o' Tc_i' Tfo_o' Tf_cL'])
legend('TLB bulk', 'Tclad outer', 'Tclad inner', 'Tfuel outer', 'Tfuel cL')
ylabel('Temperature (C)')
xlabel('Vertical Core Position (m)')
grid

figure(2)
c1f
plot(G,p2a)
ylabel('Allowable Horizontal Peak to Average Flux Ratio')
xlabel('Primary Coolant Mass Flux Through Core (kg / m^2 s)')
grid

figure(3)
c1f
plot(G,T_out)
ylabel('Average Primary Coolant Temperature at Core Exit (C)')
xlabel('Primary Coolant Mass Flux Through Core (kg / m^2 s)')
grid
Appendix C: Pump_Work Code
clear
clc

T_in = 400 + 273.15; % K
T_out = 637 + 273.15; % K
CoreHeight = 20; % m
Pitch = .015; % m
CladOD = .01224; % m
Axs = Pitch^2 - pi*(CladOD/2)^2; % m^2
Pw = pi * CladOD; % m
Dh = 4*Axs/Pw; % m
G = 1020; % kg / s * m^2
mdot = G * Axs; % kg / s

PD = Pitch / CladOD;

% Square array with s pins per side and n total pins
s = 70.0;
n = s^2;

% Buoyancy Calculation
dz_thermal_centers = 9.5; % m
grav = 9.81; % m / s^2
rho_in = 11096 - 1.3236*T_in;
rho_out = 11096 - 1.3236*T_out;
dP_B = dz_thermal_centers * grav * (rho_in-rho_out) % Pa

% Friction Calculation
mu_LB = 0.000494*exp(754.1/((T_in+T_out)/2)); % N * m / m^2
Re = G * Dh / mu_LB;
f = .184*Re^-.2;
L = 2 * CoreHeight;
De = (n * Axs *4 /pi)^.5;
rho_bar = ((rho_in+rho_out)/2);
dP_fric = f*G^2*L/(2*De*rho_bar)

% Form Losses
SumK = 25;
dP_form = G^2*SumK/(2*rho_bar)

% Pump Work
mdot_total = mdot*n;
dP_total = dP_B -dP_fric-dP_form
efficiency = .40;
W = mdot_total * dP_total/(rho_bar*efficiency)