

Thorium Fuel Cycle for a Molten Salt Reactor: State of Missouri Feasibility Study

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Thorium Fuel Cycle for a Molten Salt Reactor: State of Missouri Feasibility Study

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Abstract

This paper was generated as part of a course on advanced nuclear fuel cycles supported through a curriculum development grant from the Nuclear Regulatory Commission. The course was graduate level and required a research component. The students in the course chose the topic of “Thorium Fuel Cycle for a Molten Salt Reactor: State of Missouri Feasibility Study.” The study consisted of developing a primer to be shared with interested parties in the nuclear community and the state. This paper was generated from this research study and the approach students took towards synthesizing the primer are annotated throughout.

The thorium fueled molten salt reactor concept is one of the most promising nuclear reactor designs currently being studied, but mostly outside of the United States. Thorium reactors are desirable due to the high availability of naturally occurring fuel. However, in order for thorium fuel to be a viable alternative to uranium, a harmonious relationship between the reactor physics and molten fluoride salt transport medium must be fully understood. Chief among these technological hurdles is the use of continuous processing of spent fuel to remove fission products while the reactor is online ^[1].

The voluminous literature on molten salt reactors mostly dates to the 1960s era. Notably, in the U.S. the Molten Salt Reactor Experiment at Oak Ridge National Laboratory was an 8 MW(th) reactor that was designed primarily to study the technical feasibility and safety of using a molten salt based fuel and coolant. In addition to demonstrating the practicality of a molten salt reactor, the Molten Salt Reactor Experiment also addressed issues of on-line refueling, fuel makeup, and salt chemistry. Towards the end of the Molten Salt Reactor Experiment, and continuing after its shutdown, research efforts focused on techniques for separation of waste products, namely protactinium ^[2]. Given the prevalence of uranium based technology in the military at the time and as a matter of politics, there was little desire in the U.S. to fund nuclear research that did not provide a direct defense-related benefit.

Today, as we aspire as an industry to reduce nuclear proliferation and build safer reactors, research efforts have shifted towards reducing the amount of ²³⁵U available, particularly highly-enriched uranium. Current power reactors use low-enriched uranium (²³⁵U content of less than 4.95%). In addition to the benefits of avoiding uranium enrichment, thorium for nuclear power production is also supported by a growing demand for clean energy both in the U.S. and abroad.

Internationally, research in thorium fuels and molten salt reactors is underway in Japan, India, China, France and to a lesser extent, the United States. Presently, international research is needed in the development of molten salt reprocessing technology to allow for the active

removal of wastes from the fuel stream, while simultaneously replenishing the fuel supply. This online fuel recycling technology could lead to a significant advantage over uranium based reactors such that a fluid fuel stream allows the reactor to be continuously re-fueled, thereby significantly reducing reactor shutdown time.

Given the future economic aspirations of the nation, and in particular the state of Missouri, it is difficult to ignore the potentially huge impact of a new thorium based economy. The expansion of current mining facilities in Missouri to extract thorium could easily be achieved as thorium is present in the waste products of several mining operations already throughout the region. Moreover, the central and strategic location of the state of Missouri along the Mississippi River naturally postures the state into being a major shipping point and railway center for mineral exports. Intrastate mining, utilities, engineering, and construction companies could benefit significantly from an establishment of a thorium industry. Missouri is perfectly poised geographically, economically and academically to nurture next generation technologies for energy independence of the state and the nation as a whole.

1. Introduction

A feasibility study of thorium as a nuclear fuel in molten salt reactors for Missouri became a project for graduate students in engineering. This was a multifaceted assignment wherein students had to research, present and co-write individual sections of a primer. This project motivated students to work with industry, engineering, legal and economics terminology. The multidisciplinary aspect of the project was enriching to students who harbored different backgrounds and talents. The first stage of collaborative research and group presentations began with the history and background of potential reactors to use with thorium as nuclear fuel component. The students wanted to choose a suitable reactor. The students needed to identify past thorium use, failures and successes but also some modern advances in the area. Some students were more inclined to reactor research while others found statistics and other interesting history. Each student meeting consisted of summaries of findings by all members, recap, discussion and organization.

1.1. Nuclear Energy for Power Production

Following World War II, the United States government began encouraging the development of nuclear technologies mostly for peaceful civilian purposes, and in 1946 Congress created the Atomic Energy Commission (AEC). By the end of the 1940s, the AEC started four loosely related projects: a submarine propulsion program, a “materials testing reactor,” an experimental advanced reactor at Argonne National Laboratory, and an experimental project at Knolls Atomic Power Laboratory^[3]. Following authorization by the AEC, Experimental Breeder Reactor I was built at a remote site in Idaho and on December 20, 1951 it became the world’s first reactor to generate nuclear power. Throughout the 1950s, nuclear research was focused on demonstrating commercially viable nuclear energy^[4].

A reactor designed to use ordinary water to cool the reactor core during the chain reaction was design and built in Shippingport, Pennsylvania. Interestingly, the Shippingport reactor was

based on a reactor design that Westinghouse Electric Corporation had completed for the U.S. Navy's nuclear submarine program, and in 1954 the Joint Committee rewrote legislation to allow private ownership of reactors under an AEC license. From this point, light water reactors (LWRs) were becoming the industrial norm, and soon there was little to no room for any other reactor technology that had been developed^[3]. The Shippingport LWR was the first commercial electricity generating plant powered by nuclear fuel and it achieved its full design power in 1957. At the time LWRs were the most advanced and available design for nuclear power plants. The private industrial sector was more involved in developing LWRs after Shippingport became operational while the federal sector tended to focus on development of other reactor design, particularly for military use^[4].

1.2. The Trend towards Light Water Reactors

The Shippingport LWR was comprised of an intricate system of pipes, valves, and pumps which allowed circulating water to moderate neutrons and transfer heat from the reactor core to a steam turbine to produce electricity. The uranium fuel rested in a steel chamber through which high pressure water circulated. This design, dubbed a pressurized water reactor (PWR), was marketed by the Westinghouse Electric Corporation^[3]. Based on a similar design, the second U.S. reactor to produce electricity also used ordinary water as a neutron moderator and coolant. However, instead of maintaining the water in the reactor core under high pressure, it was allowed to boil off into steam thus operating at lower pressure than PWRs. This design was dubbed a boiling water reactor (BWR) and was marketed by General Electric. Although different in design, the PWR and BWR both fell under the category of LWRs^[3]. By the mid-1970s, LWR technology dominated the world market for nuclear power plants. By this same time, Westinghouse and General Electric, along with their subsidiaries, were winning more than two-thirds of the non-socialist country bids for nuclear power plants^[3].

1.3. Thorium in Light Water Reactors, the Renaissance

Thorium, rather than uranium, as a fuel for power reactors was genuinely considered in the U.S. during the 1960s and 1970s. While the industrial market continued to support uranium based LWRs, some thorium based technologies showed promise as a potential competitor. As energy demands rise along with commercially produced radioactive waste and as current uranium deposits continue to diminish, a shift towards new reactor designs and fuels is inevitable.

Current and next generation LWR technology, as applied to thorium-based nuclear fuel, have been investigated in recent years^[5]. The feasibility of a $^{232}\text{Th}/^{233}\text{U}$ cycle as a replacement for conventional LWRs running uranium-based fuels were majorly considered in light of the concern over growing plutonium stockpiles^[6]. Primarily, these designs have been intended for PWRs using either a homogeneous or heterogeneous fuel design. In the case of homogeneous fuel, solid fuel ThO_2/UO_2 pellets were being studied at Shippingport as early as the late 1970's and also at the 285 MW(e) Indian Point reactor in Buchanan, New York, which was commissioned in 1962 and ran until 1980. India's 300 MW(e) Advanced Heavy Water Reactor (AHWR300-LEU) runs on 20% thorium oxide fuel. The heterogeneous fuel design, such as a seed/blanket approach where uranium and thorium fuel parts are spatially separated, have also been developed internationally^[6]. These designs were initially intended for retrofitting existing PWR fuel assemblies without developing new reactor designs entirely. In addition, to LWRs,

collaboration between Purdue and Brookhaven National Laboratories has focused on thorium utilization in a tight pitch BWR lattice ^[6]. Importantly, these studies show potential for converting the current U.S. reactor fleet to utilizing solid thorium-based fuels ^[6].

1.4. Thorium in High Temperature Gas-Cooled Reactors

Thorium-based fuels have been somewhat successfully utilized in high-temperature gas-cooled reactors (HTGRs) in the past. One of the successful examples was the U.K.'s experimental prismatic 20 MW(th) DRAGON HTGR ^[7]. The DRAGON project began in 1959 and was terminated in 1976 because of political rather than technological circumstances. However, the project was a technical success from a design perspective given the successful implementation of thorium fuel with the DRAGON HTGR.

Similar success was achieved in Germany ^[6]. Two pebble bed HTGRs, the Arbeitsgemeinschaft Versuchsreaktor (AVR) prototype 15 MW(e) reactor and the Thorium High Temperature Reactor (THTR-300) 300 MW(e) reactor were successfully operated using TRISO fuel with thorium and uranium until the late 1980s after which both were decommissioned. The U.S. had fewer HTGRs that operated successfully that were eventually decommissioned, namely the prismatic Peach Bottom Unit 1 40 MW(e) experimental reactor in Pennsylvania and the commercially-operated Fort St. Vrain 330 MW(e) in Colorado which used a combination of fertile thorium and fissile uranium microspheres in a graphite matrix. In light of the performance of past gas reactors with thorium in tristructural-isotropic fuel (TRISO) or fuel rods, HTGRs have demonstrated considerable adaptability to different fuel cycles without the need for major changes to the active core design and main plant components ^[6]. Japan and China have each recently implemented domestic thorium TRISO-fueled reactors.

1.5. Thorium in Molten Salt Breeder Reactors

During the 1960s, Oak Ridge National Laboratory (ORNL) constructed the 8 MW(th) Molten Salt Breeder Reactor (MSBR). The MSBR had a graphite-moderated core with thorium fuel in a flowing molten fluoride salt. Unlike the other basic reactor formats previously described, the MSBR used a liquid fuel format in which the fuel became critical only when it was in the graphite. The molten salt served as both fuel and primarily coolant. Both fissile and fertile materials were dissolved in the molten salt. The MSBR generated valuable data on physical, chemical and neutronic properties of molten salts but also successfully tested the use of thorium-based molten fuel. The reactor tested different fuels including ²³⁵U, ²³³U, and plutonium. Recently, China, France, Japan but to a lesser extent the U.S. have all further researched MSBRs, with the thorium cycle presenting itself as a highly viable option, Japan and China are currently developing programs with thorium fuel using molten salt technology. China plans to build two prototype thermal MSBRs by 2017 and the project is valued at \$350 million. Japan's 200 MW(e) FUJI molten salt thermal breeder reactor has gained support from the U.S. and Russia ^[2, 6, 8].

1.6. Rationale for a Thorium-Based Fuel Cycle for MSBRs

From the mid-1950s to mid-1970s, thorium based fuel cycles were of considerable interest worldwide as a means to supplement uranium reserves without contributing to plutonium buildup. The feasibility of thorium in LWRs, HTGRs, and MSBRs have been basically

demonstrated but historically underestimated. The use of thorium as a liquid fuel in a MSBR is arguably one of the most attractive reactor concepts due to the possibility of a closed fuel cycle, no fuel melt-downs, shorter-lived radioactive waste, proliferation reducing aspects of the fuel, the natural abundance of thorium, and the favorable economics. These points are explored in detail and are the purpose of this research. The other main purpose of this research is to appraise the significant economic, industrial and scientific merits of thorium as a source of nuclear energy for Missouri.

2. Basics of Thorium

Once the core of the project was identified the students shifted towards the research and information gathering stages. The students found historical information but also basic radiochemistry and chemistry knowledge about thorium. While the history of thorium had international and national events, in the following sections students also had an opportunity to employ local resources to gather information on thorium mining in Missouri. State office websites as well as alumni contacts were extremely valuable. Public information about the status of local Missouri mines were often mentioned in the newspaper and online sources over the years.

Thorium, a transition metal and the 90th element on the periodic table, occurs naturally in thorite and monazite mineral crystals found throughout the world ^[9]. It was initially discovered in 1828 by Jons Jakob Berzelius, a Swedish chemist, who named the newfound substance after Thor, the Norse god of thunder and weather. Thorium has numerous non-nuclear applications, including the coloring of ceramic glazes and the tinting of some ophthalmic lenses. Additionally, thorium is known to enhance the burning capabilities of lantern mantles and welding rods, and is used as an alloying metal in some aerospace materials. Several decades ago, thorium was also utilized as a diagnostic imaging solution in hospitals, but has since been replaced by other more specialized radioisotopes ^[10]. In 1898, with only two months between their discoveries, Gerhard Carl Schmidt and Marie Curie each independently determined that thorium was radioactive. The half-life of ²³²Th is about 14 billion years and undergoes alpha decay with gamma emission.

Trace amounts of thorium are present almost everywhere. Present in rock, water, and soil, thorium is taken up through plants and is present in the food chain. The radiation from these minute traces is indistinguishable from background, and therefore poses no threat to human health. Monazite, thorianite, and other materials can be mined for their thorium content. In mining operations, inhalation of thorium is the greatest risk for long term exposure, similar to uranium mining ^[10].

3. Mining in Missouri

Thorium is found in a variety of rare earth mineral ores, including monazite, thorite, and thorium phosphate. In the course of mining monazite, thorium along with other heavy metals and rare earth elements (REEs) can be extracted as byproduct. Monazite is a major rare earth-bearing mineral ore with the following composition (Ce, La, Nd, Th)PO₄. The existing

concentrations of thorium in currently-mined rare earth mineral ores are extremely beneficial because they will allow for the use of pre-existing mining facilities and infrastructure with little modification, thereby reducing the costs associated with expansion of thorium mining in the U.S.^[11]. The long-term demand for monazite is predicted to increase due to the mineral being in abundant supply and the ease of extracting valuable byproducts.

As shown in Table 1, although the statistics vary from study to study, the U.S. generally has one of the largest known reserves of thorium in the world. Currently, China and Russia are trying to determine the extent of their thorium reserves. It is anticipated that these countries will have significant amounts due to the high concentration of REEs found in igneous and metamorphic rocks. Eastern Asia, India and Eastern Europe are projected to be key suppliers of rare earth minerals as both the cost of exploration and domestic demand increase in China.

Table 1. 2011 Estimates of world thorium reserves^[11].

Country	Reserves (tons)
United States	440,000
Australia	300,000
India	290,000
Canada	100,000
South Africa	35,000
Brazil	16,000
Malaysia	4,500
Other countries	90,000
Total	1,275,500

3.1. Thorium in Missouri

Compared to other states in the U.S., Missouri has been historically rich in rare earth minerals and possesses a unique and advantageous opportunity to lead the forefront of thorium mining and production in the country. First-to-market will be crucial for the state. Being centrally located in the country, situated nearby the Mississippi River, having many rail and truck transportation outlets and enjoying a long history of mining and industrial operations, Missouri can be a key player towards commercialization of thorium fuel. Missouri is also a leader in nuclear research and power production with the successfully operated Ameren's Calloway nuclear power plant located in Fulton, Missouri and two experimental research reactors owned by the University of Missouri system, including the University of Missouri Research Reactor (MURR) in Columbia, Missouri being the largest University research reactor in the country. The mining industry in the state of Missouri could easily benefit from expansion of their current mining operations to include the extraction of thorium for nuclear applications. In Missouri, further exploration of thorium is warranted in the southeastern part of the state due to the presence of iron and lead deposits.

3.2. Mining for Thorium in Missouri

Historically, Missouri has majorly supported two main industries based on its geographical resources: farming and mining of rare earth elements. Mining in Missouri includes arsenic, cadmium, cobalt, copper, lead, nickel, and zinc as the primary trace elements associated

with sulfide minerals. Figure 1 shows the present mining operations in Missouri primarily for the heavy element lead. Given Missouri's nuclear expertise, extensive mining infrastructure in the state, and the presence of thorium as a by-product in existing resources, it is logical to add thorium processing lines to existing heavy metal extraction facilities. Additionally, regions of the state where vast amounts of thorium is found would stand to gain a significant economic benefit. The Pea Ridge Mine in Washington County, Missouri was one of the largest iron ore mines in the world and the only deposit with confirmed rare earth element concentrations ^[12]. Closed in September 2001 due to flooding, the mine is located near Meramec State Park about 16 miles from Sullivan, Missouri. As the competition with China for REE-made electronic components increases, new owners of the Pea Ridge mine in 2012 were discussing to reopen the mine for REEs and also for extracting thorium ^[13]. In 1984, the mine produced 1.3 million tons of ore pellets and the remaining reserves are estimated at 100 more years, making this mine one of the richest ore bodies in the world. Missouri has three world-class lead/zinc areas in the southwest region referred to as the Southeast Missouri Lead District, as shown in Figure 2.

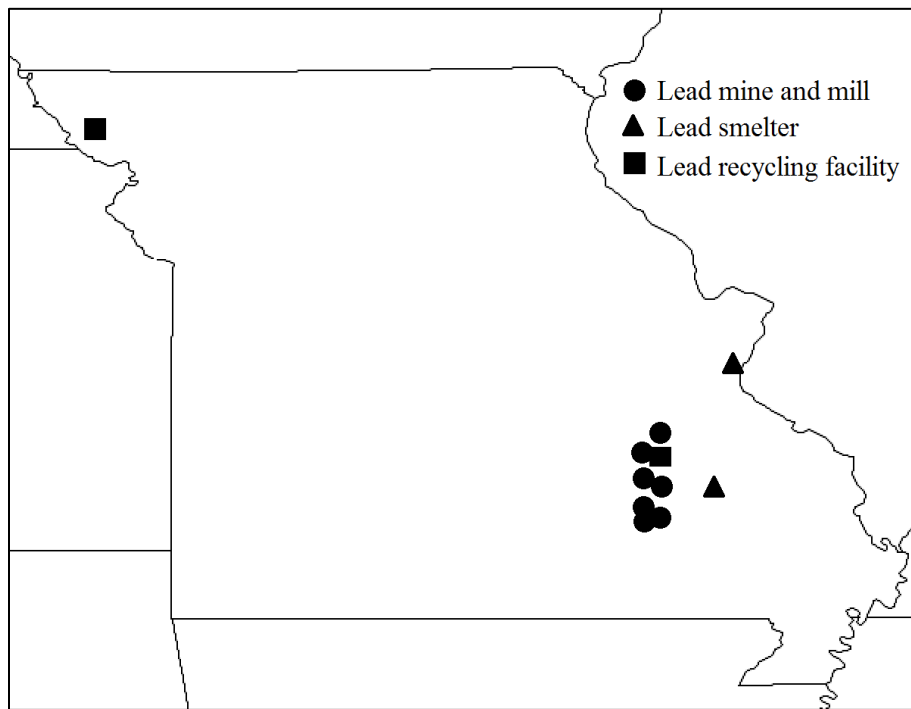


Figure 1. Missouri lead mining industries ^[14].

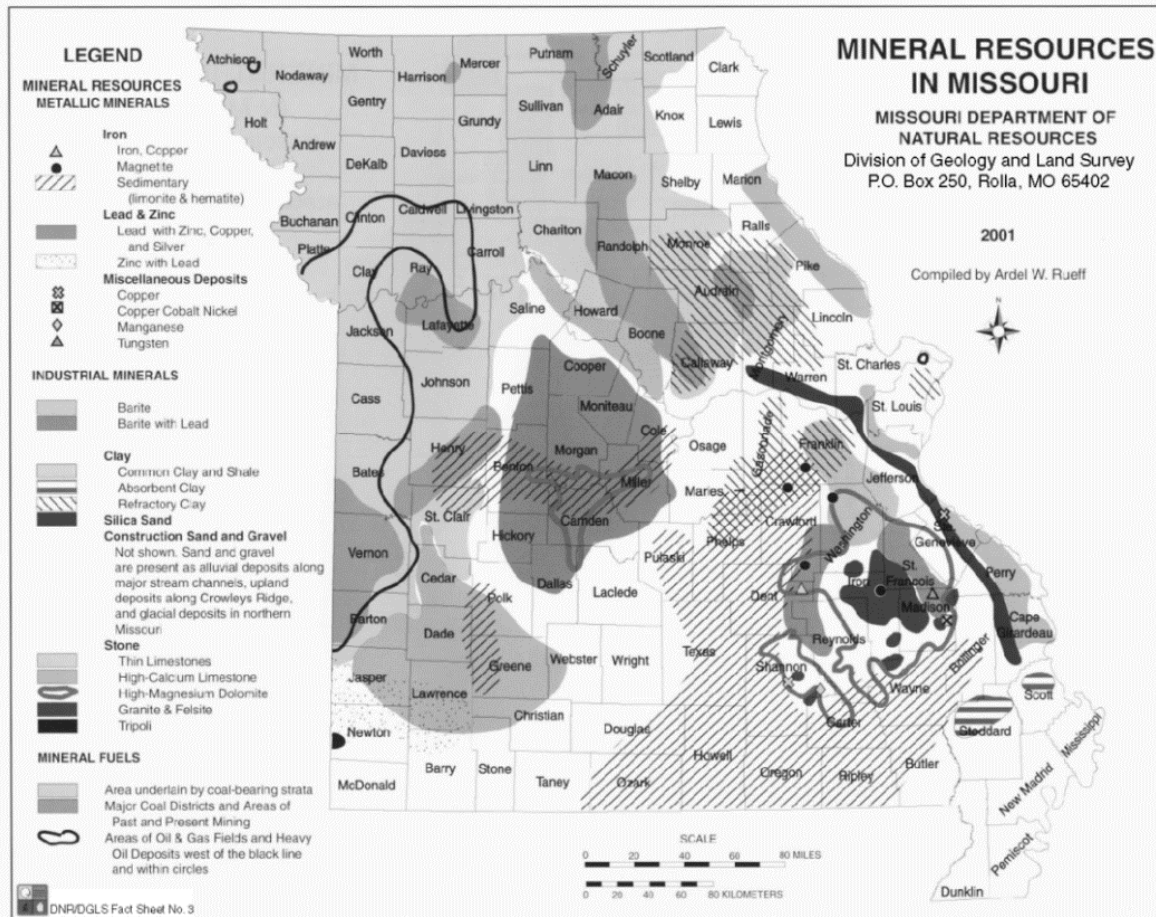


Figure 2. Missouri natural resources map^[14].

3.3. Potential Challenges for Thorium

This report proposes further exploration throughout the state of Missouri to fully assess the extent of thorium deposits in existing mining operations. REEs that are potentially rich in thorium are also in high demand as China has significantly decreased export of materials due to costs but also due to increased in-land demand^[15]. Furthermore, because mine tailings are considered hazardous waste, special precautions will be necessary to ensure the safe storage of the tailings. The Missouri Department of Natural Resources has already established guidelines and inspection protocols for the control of these wastes^[12]. In addition, the content of the tailings for rare earth metals including thorium should be evaluated.

4. Thorium Molten Salt Reactor Design and Engineering

The students approached the following section and Section 6 as the engineering sections of the primer. The technical aspects of the reactor and fuel were researched and the molten salt reactor was proposed by the students. The technical section had several parts including the choice of fuel format, choice of driver fuel, breeding ratio, reactor size as well as a survey of available advanced chemical technology to reprocess the fuel. At this point of the project,

students are well-informed on the issues and gaps in the research and can make recommendations for further work and study.

The Molten Salt Reactor (MSR) is one of the most attractive reactor concepts included in the family of Generation IV reactors due to its inherently safe design, innovative liquid fuel format, online fuel reprocessing capabilities, short doubling times and ease of small modular construction. MSRs are a type of high-temperature, salt-cooled reactor used for producing electricity, burning actinides and producing hydrogen but also for breeding fissionable species, such as the MSBR^[16]. MSRs have a strong negative temperature coefficient and negative void coefficient, so as the temperature in the core rises the density of the fuel increases and moves away from the core^[17]. The reactor concept is distinguishable from modern reactors due to a complex balance achieved between reactor physics and chemical fuel reprocessing technology. Historically, an impressive breeder demonstration was performed at Shippingport in the late 1970s and early 1980s using a $^{232}\text{Th}/^{233}\text{U}$. Fissile ^{233}U was produced from breeding ^{232}Th fuel. The thorium nuclear fuel required no enrichment but would have required reprocessing. This was the only U.S. nuclear research program using ^{233}U as the fissile seed material, and it created a great incentive for future research^[1]. Few would have guessed that this underfunded research would resurface 30 years later for actualization and hopefully, commercialization.

From the perspective of thorium advocates, MSBRs have clearly become a top choice for nuclear fission reactors and a key concept for Next Generation Nuclear Plants with hydrogen production facilities (NGNPs). The technological and engineering advantages of a single-fluid thorium molten salt breeder reactor include fluoride liquid fuel, the ability to breed fissionable species at a constant and high ratio, online chemical reprocessing of nuclear fuel, better waste management and the potential for small modular construction. In addition, “[t]he use of a $^{232}\text{Th}/^{233}\text{U}$ breeder reactor cycle results in high-level waste with a very low actinide content because as neutrons are added to the thorium the various fissile isotopes that are produced (^{235}U , ^{239}Pu , etc.) tend to fission”^[16].

4.1. Advantages of a Thorium Fluoride Liquid Fuel

Currently, all nuclear power plant operations in the U.S. use solid fuel (uranium- or plutonium-based) in either a PWR or BWR. A thorium-based fuel cycle bypasses the need for an enrichment facility but would still require the use of a uranium- or plutonium-based fuel or perhaps, transuranic waste (TRU) as a driver fuel. Driver fuel (also known as start-up fuel) is necessary in a breeder reactor to build up an initial inventory of fissile species, in this case ^{233}U .

There are numerous technological and engineering advantages of an MSBR using thorium-based fuel. In a thorium MSBR, the primary coolant and nuclear fuel are comprised of a molten salt mixture^[16]. The molten salt composition depends on the type of reactor and operation it performs but usually the optimized salt composition is around 71.7 ^7LiF , 16.0 BeF_2 , 12.0 ThF_4 and 0.3 UF_4 mol%^[1, 16]. Fluoride liquid fuels have excellent chemical, neutronic and thermodynamic properties. They have a low melting point of about 360 to 459°C and a very high boiling point of about 1400°C^[1, 17]. These properties are quite advantageous since the reactor system is projected to operate between 500 and 705°C, leaving a large margin to avoid salt freezing^[16]. Additionally, the liquid fuel has a low vapor pressure above the liquid during operation^[18]. The liquid fuel saves pumping costs as there is no need to pressurize the fuel.

The lithium fluoride in the fuel salt mixture is 99.9999% isotopically-enriched to ^7Li to avoid tritium production from $[n, \alpha]$ reactions with ^6Li . The fluoride mixture has a low thermal neutron capture cross section of 40 mb for ^6Li and 9 mb for ^9Be , and experiences no radiolytic decomposition when irradiated ^[1, 17]. Molten salts have a great solubility for fissile (uranium and plutonium), fertile (thorium) and fission isotopes. Lithium fluoride salts are excellent coolants, having a 25% higher volumetric heat capacity than pressurized water and about five times the heat capacity of liquid sodium ^[16, 17].

4.2. The Single-Fluid Thermal Breeder Reactor

Depending on the operation desired, there is a choice of several thorium MSBR models. MSBRs generally come in two basic formats: single-fluid and double-fluid core models. There is also the major choice of operating with a hard or soft neutron spectrum. For instance, France has a fast MSBR which requires a high driver load of about 5.5 tonnes of uranium fuel/GW(e) as opposed to a thermal breeder which would require about 1500 kg of uranium driver fuel/GW(e) ^[17]. Currently, molten salt technology in the U.S. is mostly focused on utilizing the molten salt as a coolant for high temperature solid fuel thermal reactors using TRISO fuel rather than considering molten salt for fuels research.

The single fluid MSBR has the fissile, fertile and fission isotopes dissolved in a convenient, single molten salt mixture. The double fluid MSBR has two separate salts, namely, a fuel salt and a blanket salt that contains fertile thorium isotopes. In a double-fluid reactor, it is challenging to keep the blanket and fuel salt spatially separate and at a constant reactivity due to graphite swelling and shrinking caused by irradiation, which eventually causes “plumbing problems” ^[1, 17]. On the other hand, a single-fluid reactor has a simpler core design. Both models have their advantages and disadvantages and the suitability of each reactor must be determined on an individual basis. However, the single-fluid MSBR is an attractive model because of the simplicity of having a single salt mixture, energy savings of at least 10% from not having separate heat loops for blanket salt and the benefit of five years of successful operation of the ORNL 8 MW(th) single fluid thermal breeder reactor in the 1960s ^[17].

A thermal or epithermal reactor is most suitable for a thorium-based fuel cycle as demonstrated by the ORNL breeder reactor. A thermal neutron flux is easier to control although an epithermal flux would help reduce the build-up of ^{238}U . Further, at low average neutron fluxes, in-situ breeding is much better than for uranium-based fuel ^[17]. A fast breeder reactor (FBR) also has the disadvantage of requiring more driver fuel at startup and reloading, and it generates a larger volume of waste and longer-lived radioactive waste than a thermal breeder as illustrated by the data in Figure 3 ^[17, 19]. Additionally, a fast reactor would probably require a higher initial economic investment. Both the FBR and the thermal MSBR generate less TRUs, namely Np, Pu, Am, Cm, than a typical PWR. All fission products (FPs) waste from fission of uranium tends to diminish after a few hundred years.

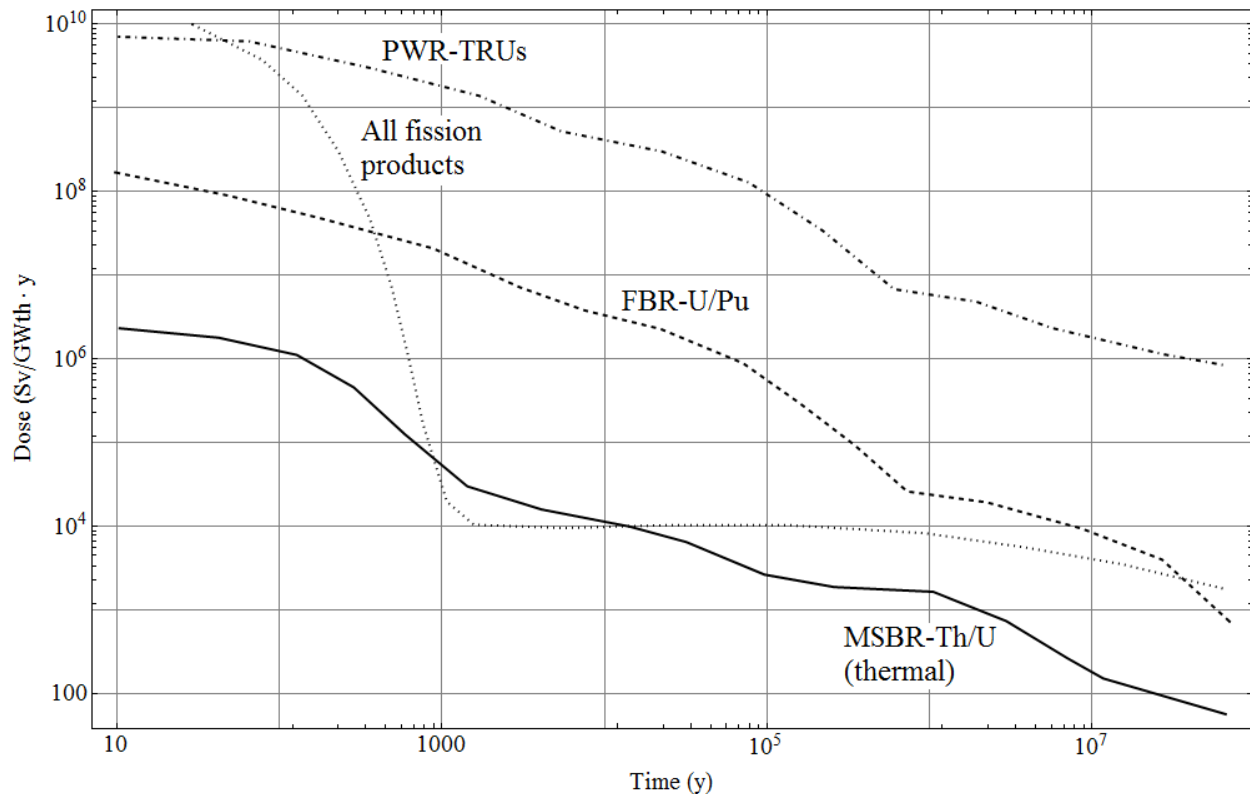


Figure 3. Comparison of the radiotoxicity of nuclear waste from thermal to fast flux reactors ^[20].

5. Thorium versus Uranium

The radiochemistry research was approached by the students in a similar way as the technical research on the reactor. Engineering students were inclined to these sections but the local history research and background information was valuable to the students' overall understanding of the issues.

5.1. Advantages of Thorium as a Nuclear Fuel

Below are some basic reasons for the selection of thorium over uranium-based nuclear fuel cycle ^[6].

- Increased fissile resources by breeding ²³³U from thorium
- Improved fissile fuel utilization in thermal reactors
- Significant reduction in ²³⁵U enrichment requirement
- Decreased production of plutonium and other transuranic elements compared to a uranium fuel cycle
- Advantageous neutronics and physical properties of thorium-based fuel
- Opportunities for long-term local economic growth in Missouri
- Significant reduction in weapons proliferation risks

From a neutronics perspective, ^{233}U has a higher neutron yield per neutron absorbed than either ^{235}U or ^{239}Pu in a thermal neutron flux. The average number of neutrons produced per absorption of neutron (the η factor), is 2.27 for ^{233}U in standard PWR compared to 2.06 for ^{235}U and 1.84 for ^{239}Pu . In fact the eta factor (η) of ^{233}U is greater than 2.0 over a wide range of neutron spectrums, allowing thorium to be used in a wide range of fluxes. Therefore, ^{233}Th is a theoretically viable and ideal fuel for nuclear reaction. In addition, ^{233}Th has a higher thermal neutron absorption cross-section (7.4 barns) than ^{238}U (2.7 barns), which makes it an outstanding fertile fuel. Additionally, the conversion of ^{232}Th to ^{233}U is much more efficient than that of ^{238}U to ^{239}Pu . Another advantage is that the capture cross-section of ^{233}U is much smaller than that of ^{235}U and ^{239}U for thermal neutrons [6].

From a non-proliferation perspective, ^{233}U is produced in the thorium cycle, which is crucial because the half-life of ^{232}U is only 73.6 years. This results in the daughter products having a very short half-life, and some, like ^{212}Bi and ^{208}Tl emit strong gamma radiations: 0.7-1.8 MeV and 2.6 MeV, respectively, which increases the difficulty of manufacturing nuclear weapons [6]. Moreover, very little if any ^{239}Pu is expected to be produced and virtually impossible to separate.

For instance, in an Accelerator Driven subcritical assembly System (ADS) with Energy Amplifiers (EA), using thorium as breeding fuel would minimize the radiation risk of waste and the risk of proliferation at the same time. The waste from such thorium-fueled EAs is approximately 30 times less toxic than that of uranium-based fuel for the first 30,000 years of cooling time [6]. The thorium-based cycle will produce shorter-lived radioactive waste for storage and decay compared to a $^{238}\text{U}/^{239}\text{Pu}$ -based cycle. This is illustrated in the fission yield graph of Figure 4. A much smaller quantity of plutonium and long-lived minor actinides (Np, Am, and Cm) are formed as compared to the $^{238}\text{U}/^{239}\text{Pu}$ fuel cycle, for instance. Therefore, thorium has a greater advantage as a nuclear fuel from a waste management perspective.

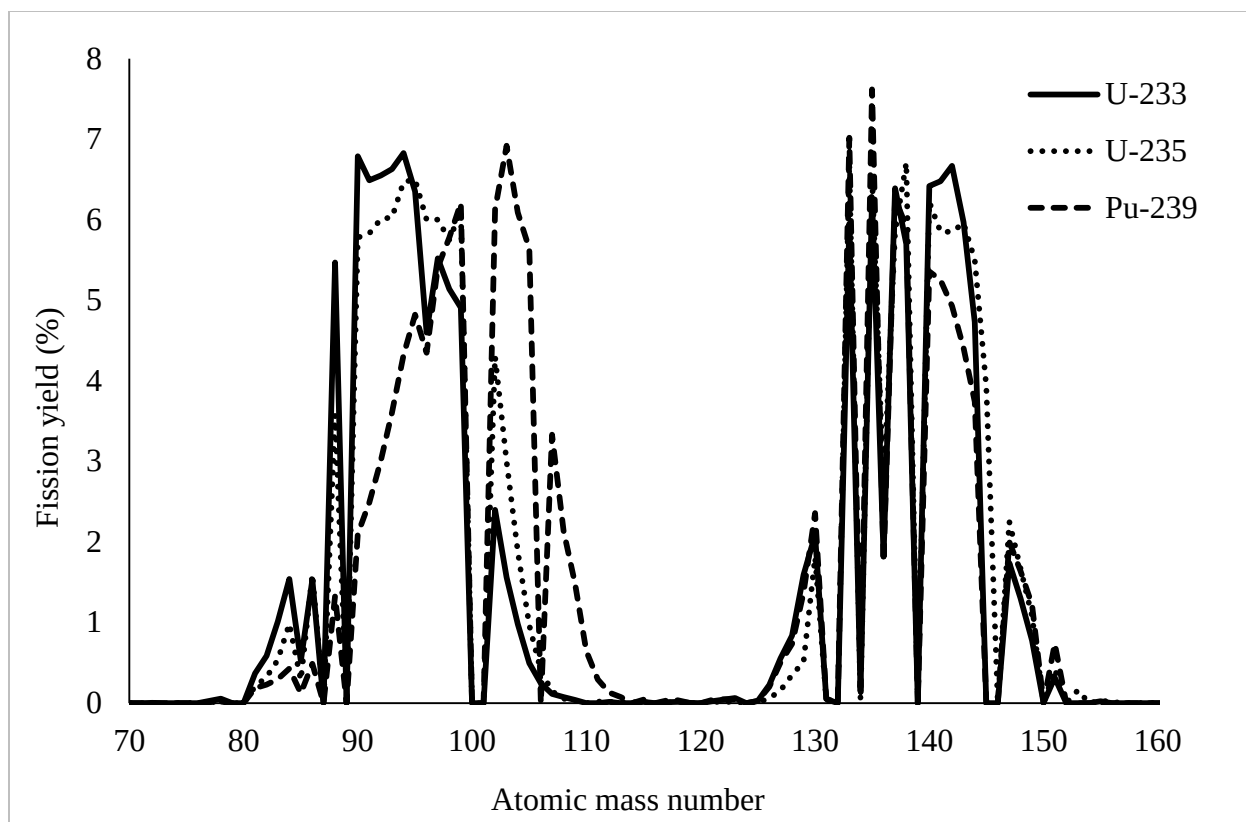


Figure 4. Fission yield versus atomic mass number for various fuel cycles ^[21].

5.2. Protactinium Radiochemistry

In the conversion chain of ^{232}Th to ^{233}U , ^{233}Pa is formed as an intermediate. ^{233}Pa has a relatively longer half-life (~ 27 days) as compared to its counterpart ^{239}Np (~ 2.4 days) in the uranium fuel cycle. During this period of time, ^{233}Pa would absorb a significant amount of neutrons due to its high thermal neutron cross-section, and decay to ^{234}U instead of fissile ^{233}U . As a consequence, it is necessary to have a storage and decay time of at least 12 months (more than 10 half-lives of ^{233}Pa), prior to reprocessing, in order to complete the decay of ^{233}Pa to ^{233}U and avoid loss of any ^{233}U fissile material.

6. The Single-Fluid Molten Salt Reactor: System Description

Figure 5 shows a flow diagram of a basic single-fluid thermal molten salt breeder reactor system with potential process heat applications ^[16]. Molten fuel salt flows between the reactor core and primary heat exchanger which are housed together in a containment building. In a thermal breeder reactor, the core is moderated with unclad hexagonal graphite fuel elements with a hole bored through the center called a “salt channel” as shown in Figure 6 ^[16]. The fuel salt flows through the graphite salt channels and becomes critical in the graphite and then flows to the primary heat exchanger. The core is constructed of stacked hexagonal graphite fuel elements and there are also two graphite reflectors.

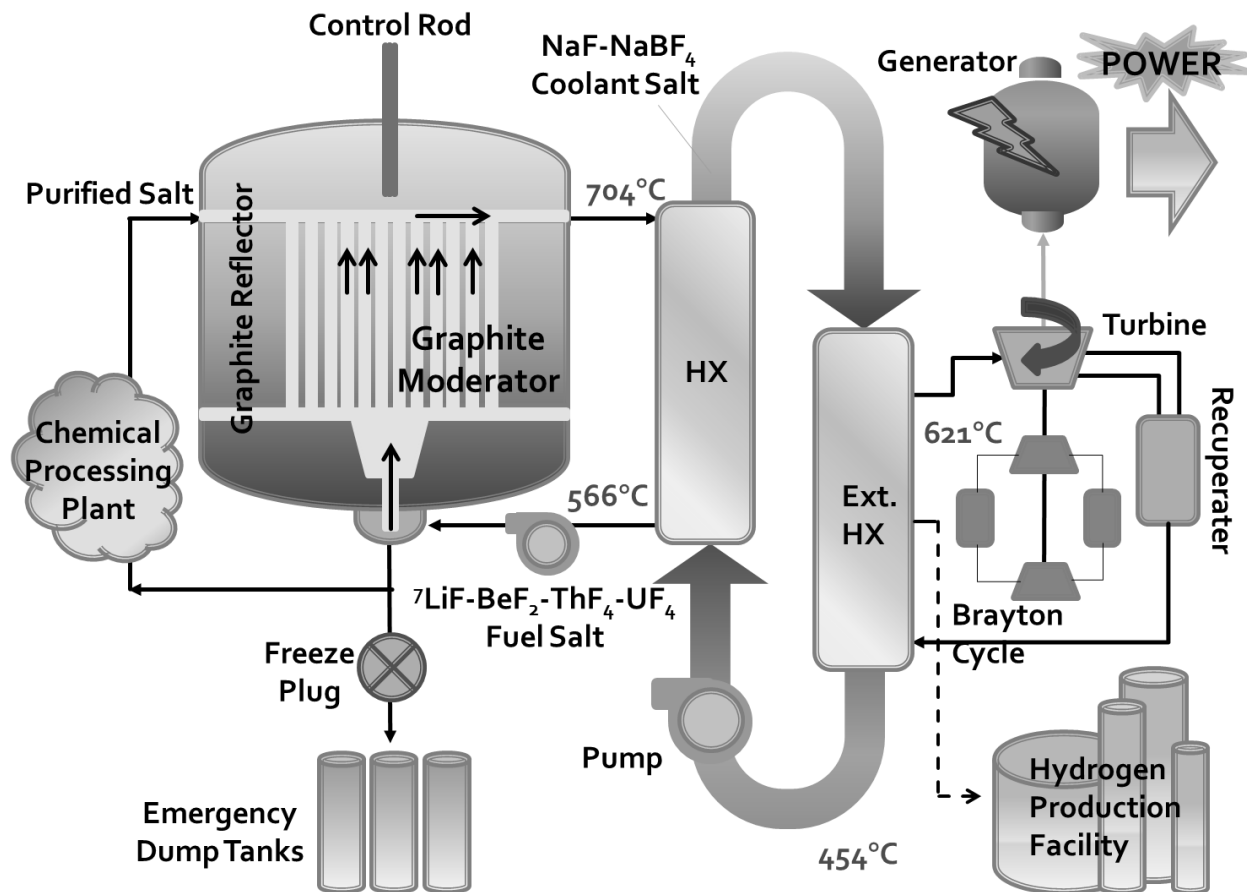


Figure 5. Single-fluid thorium molten salt thermal breeder reactor ^[16].

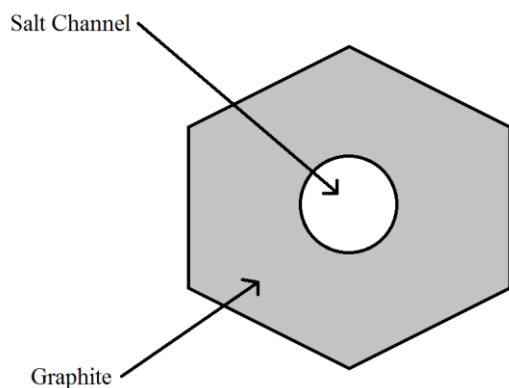


Figure 6. Cross-sectional view of unclad hexagonal graphite fuel element.

The MSBR is housed in a low-pressure vessel constructed of HastelloyN® several inches thick ^[16, 19]. A secondary fluoride salt, NaF-NaBF₄ transfers heat to a high temperature Brayton cycle to produce electricity ^[16]. The use of a helium or nitrogen Brayton cycle, as opposed to a

Rankine cycle, avoids the use of steam. Further, the higher temperature of the Brayton cycle assures that salt freezing will not occur in heat exchangers ^[16].

The reactor is operated at steady-state with no change in reactivity with time. In fact, the reactivity is much easier to control in a single-fluid reactor ^[17]. The core inlet temperature is projected to be about 566°C and the outlet is about 704°C ^[16]. Heat may also be passed to a hydrogen production facility and although a higher core outlet temperature of about 850°C is required, this can certainly be achieved with materials optimization ^[16]. In the case of an accident, a freeze plug is designed to melt allowing the fuel to drain into critically-safe, passively cooled dump tanks ^[16, 18, 19]. These tanks are also further backed by a main drain for the containment building.

Spent fuel is constantly reprocessed at an on-site chemical processing plant that removes fission products and separates the neutron poison ^{233}Pa . Clean salt is returned to the reactor core along with fresh fuel. Current research and development in MSR technology is underway only by a few groups across the E.U., U.S. India, China and Japan. More research is needed to optimize the online fuel processing chemistry so that the fuel can be continuously cleaned and at a high rate to maintain decent breeding ratios (typically around 1.06) and also to minimize neutron losses due to ^{233}Pa absorption ^[17]. The ability to reprocess the irradiated thorium fuel is the main feature to the entire reactor concept but it is important to note that it was never fully conceived by ORNL before funding cuts were underway ^[1].

Chemical separation technologies have vastly improved over the last thirty years in other fields and so has nuclear waste management. With current resources it seems very likely that efficient, high volume online fuel reprocessing will soon become a reality. The separation technologies under development are liquid metal extraction, electrochemical separation (anodic dissolution), fused salt volatilization, and gas extraction. Since gas strippers and anodic dissolution processes are non-aqueous, there is the possibility of building a compact facility ^[1, 17].

6.1. On-Site Spent Fuel Reprocessing

Figure 5 shows that a chemical reprocessing facility could be built adjacent to the reactor building. In the reprocessing facility radioisotopes must be separated chemically and sequentially but also at very high volumes to maintain high breeding ratios in the core. The decay heat emitted by isotopes during fuel cleaning is very favorable during the reprocessing stage, easing the separation of isotopes ^[17]. The online reprocessing is a crucial step to the whole system. Breeding is promoted by keeping a continuous and efficient fuel reprocessing regiment. A high volume of both chemical and radiative waste is generated. Figure 7 shows a basic spent fuel reprocessing facility to separate ^{233}Pa to a semi-batch decay unit where it decays to ^{233}U which is returned to the reactor core.

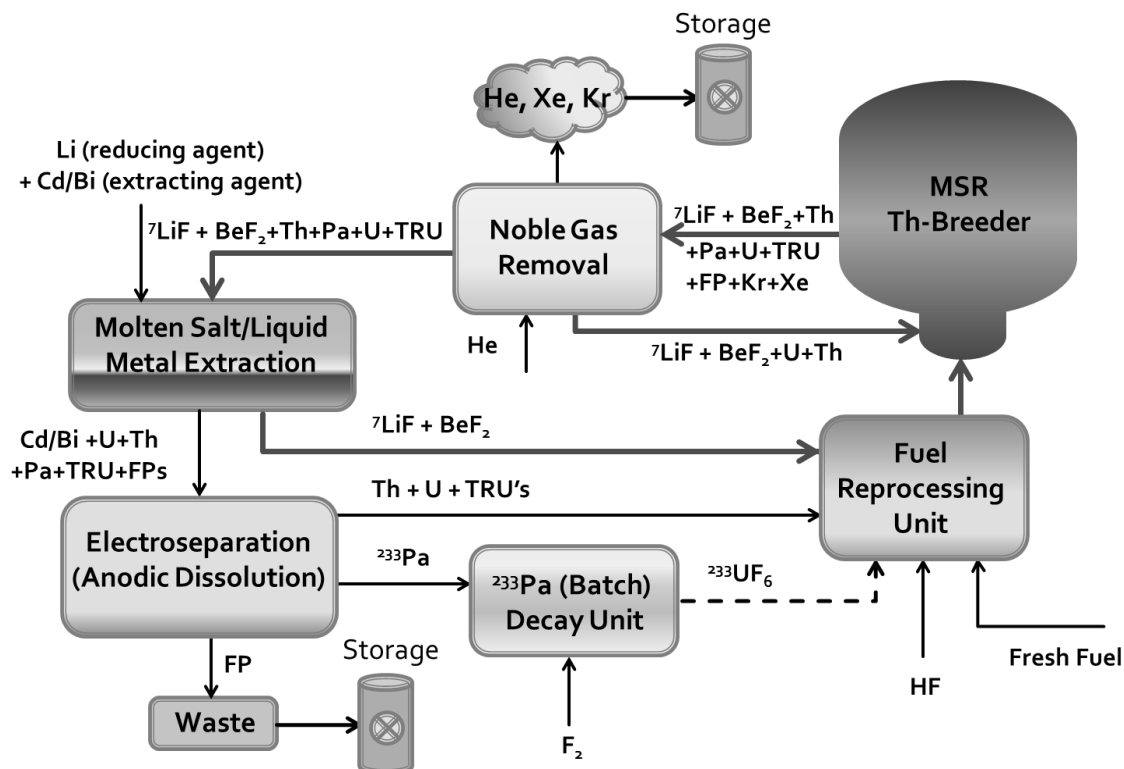


Figure 7. Basic flow diagram of an on-site spent fuel reprocessing chemical facility ^[1].

Spent fuel from the MSBR first goes to a noble gas removal unit in which helium is used to bubble out gaseous fission products including xenon, krypton, and helium at very high rates for storage ^[1, 19]. Noble gases do not dissolve in the liquid salt like other fission products so they are removed first. Furthermore, it is important to remove them quickly to avoid any “plateout” on heat exchanger surfaces ^[17]. The heat exchangers in a molten salt reactor system, unlike other reactor systems, are exposed to a high level of radioactivity.

The spent fuel then flows to a molten salt/liquid metal extraction unit in which everything in the salt phase is transferred to liquid bismuth. Lithium is used as a reducing agent and cadmium or bismuth are used as extracting agents, each with their own recycle loops (not shown in Figure 7) ^[1, 17, 19]. The clean salt then flows back to the reactor. The extracted species then continue to the electroseparation unit in which fission products, ${}^{233}\text{Pa}$, ${}^{233}\text{U}$, and TRUs are separated very selectively and sequentially based on redox potentials. ${}^{233}\text{Pa}$, having a half-life of 27 days and being a considerable neutron poison, is sent to a batch decay unit and allowed to decay to fissile ${}^{233}\text{U}$ for about the duration of ten half-lives, making the whole process a semi-batch operation. ${}^{233}\text{U}$ is then fluorinated with fluorine gas, which makes an excellent stripper due to a high electron affinity, and then the fissile ${}^{233}\text{U}$ is returned to the reactor core ^[1, 17].

Attempts to extract ${}^{233}\text{Pa}$, ${}^{233}\text{U}$, and ${}^{232}\text{Th}$ by the HNO_3 -TBP/Kerosene solvent extraction route have not been successful. However, selective adsorption of ${}^{233}\text{Pa}$ in Vycor glass with subsequent elution, preceding the extractive separation process, has been reported to be successful. Nearly 98% of ${}^{233}\text{Pa}$ could be separated by this novel method. There are also some other technologies to extract ${}^{233}\text{Pa}$ that are currently underway, such as: adding inorganic metal

oxide precipitants, using high surface iron (steel wool), and fluoride volatilization techniques etc.^[22]

6.2. Core Maintenance and Waste Management

One drawback of a graphite-moderated core is the need for core replacement every few years due to irradiation damage. In this case, the construction of a fleet of modular units can prove very advantageous. For example, a fleet of four small modular 250 MW(th) thorium-based thermal breeder reactors can always guarantee 75% power while one out of four units is shut down for core maintenance^[23]. Further, a single-fluid MSR, as opposed to a double-fluid design, requires only that the graphite in the core is replaced. In contrast, a double-fluid reactor may require the replacement of the entire pressure vessel which must be done remotely with robotics^[17]. ^{233}U is almost always associated with ^{232}U which has a very high specific activity because of its radioactive daughters, namely ^{208}Tl , which is a hard gamma emitter. ORNL reported that ^{208}Tl was highly corrosive to reactor components built from earlier formulations of HastelloyN®^[17]. Thus, handling and processing of ^{233}U as well as reactor charging requires additional biological shielding compared to a plutonium-based breeder. This aspect certainly contributes to costs. Reprocessed thorium contains ^{234}Th and ^{228}Th , which has ^{208}Tl as a daughter product, preventing direct handling for at least tens of years^[19].

Selected fission products are removed from the molten salt and vitrified in storage units as shown in Figure 7. Disposal of fission product waste from reprocessing thorium-based fuel would require treatment similar to that of waste from reprocessed uranium-based fuels. Generally, thorium-based fuel cycles produce much less plutonium and minor actinides than uranium-based fuels but they instead generate other radionuclides such as ^{231}Pa , ^{229}Th , and ^{230}U that have a long-term radiological impact^[17-19]. Any plutonium produced remains in the salt. Lighter plutonium fissile isotopes burn out faster than ^{242}Pu , having major implications from a standpoint of reducing weapons proliferation. “The high ^{242}Pu content makes the plutonium for a MSR much less desirable than plutonium from any other reactor type for use in nuclear weapons because of its very high critical mass...”^[16].

Utilization of the $^{232}\text{Th}/^{233}\text{U}$ cycle produces several orders of magnitude less TRUs than a conventional once-through cycle and significantly less than even a $^{238}\text{U}/^{239}\text{Pu}$ fast breeder^[16, 17]. Radiotoxicity levels are less than equivalent uranium ore levels within a few hundred years^[17, 19]. MSRs are also advantageous for use in destruction of TRUs from traditional once-through reactors but TRUs may also be used as a driver fuel to build an initial inventory of ^{233}U ^[17].

6.3. The Conceptual Small Modular Single-Fluid Molten Salt Breeder Reactor

In summary, a single-fluid small modular thermal or epithermal MSR with a thorium fuel cycle could have the following design features:

- Use TRUs from once-through LWRs or PWRs to build a startup ^{233}U inventory, after which only thorium needs to be added to refuel the reactor
- Form a small fleet of four reactors which guarantees 75% power at all times, for instance, if one unit out of four was down for maintenance or core replacement
- $^7\text{LiF-BeF}_2$ molten salt fuel with 67 K safety margin on the entering temperature

- The core is inside a low pressure vessel constructed of ASME Sec. III qualified materials, such as Alloy 800H clad with HastelloyN or INCONEL
- Achieve approximately 1.06 breeding ratio with constant online fuel processing/²³³Pa removal
- Incorporate a Brayton power cycle (44% efficiency) and potential for hydrogen production
- 20-year reactor doubling time

7. Economics of a Molten Salt Reactor

Some students had taken courses in reactor and plant economics in their undergraduate design courses. They incorporated some of that knowledge to review the economic feasibility of building and relying on molten salt reactors. A few economic reports were available as well through archives from the National Labs. Here, engineering students had the opportunity to consider the real-world economics but also try to project the cost of a R&D program in the state. The students felt that preparing the economics portion of the primer well was essential to the entire primer.

Until recently, thorium-based MSRs have never been fully realized as a viable reactor concept so the economic cost projections are somewhat based on speculation and best-estimates. In 2011, the Chinese Academy of Science launched a thorium-fueled MSR R&D program which is financially-backed by \$350 million dollars for five years and plans to build prototype Generation IV MSRs. The first of two test reactors will be built in 2015 (TMSR SF1) and the second in 2017 (TMSR SF2)^[8]. Furthermore, India, which holds around 22% of the world's thorium reserves, scheduled four new fast breeder reactors to be built by 2020. These forthcoming projects may give some insight on the size of the venture.

The approximate cost of a MSBR in 1966 was \$113.6 million^[24, 25]. The modern construction cost could be estimated by updating this cost estimate with assumed multipliers and by making an inflation-adjusted estimate. Table 2 shows that the modern total direct cost could be more than \$600 million for a 1000 MW(e) MSBR without considering the indirect costs for engineering R&D, contingencies, etc. The indirect costs, also listed in Table 2, correspond to about 33% of the direct construction costs and give total MSBR venture a rough estimate about one billion dollars. The costs of materials, fabrication, inspection, transportation, installation, and testing were projected assuming there were no barriers to accessibility.

7.1. Capital Costs

The basis for the capital cost estimates are given in Table 2. The 2011 costs were increased by a multiplier based on the increase in construction materials and labor costs from 1966 to 2011. A multiplier was based on considerations of the 2011 Consumer Price Index and risk-free inflation calculations. The indirect costs for 2011 include the labor requirements and labor services, namely construction services, home-office engineering services, and field-office engineering services. The direct costs for 2011 are as follows:

- Land and Land Rights: The cost of land was not included in the total direct construction costs but treated as a separate estimate as land values keep fluctuating in today's market

- Structures and Improvements: The primary and major structure in the MSBR plant is the reactor containment building which will incur about 50% of the account. The cost of the fuel reprocessing facility has been included here and also in construction costs, although R&D costs were not projected. While layouts of the internal areas of the MSBR plant would differ, the cost of the plant will probably be competitive with that of the PWR. The estimate was further adjusted for lifetime storage of radioactive waste on-site, including provisions to facilitate decommissioning.
- Reactor Plant Equipment: Reactor vessel and internals, control rods, shielding and containment, heating-cooling systems and vapor-suppression system, moderator and reflector, and reactor plant crane are some of the major equipment estimate herein ^[24].
- Turbine Plant Equipment: Turbine-generator units, circulating-water system, condensers and auxiliaries, central lube-oil system, turbine plant instrumentation, turbine plant piping, auxiliary equipment for generator, and other turbine plant equipment are considered in the account ^[24, 25].
- Electric Plant Equipment: This account covers the costs of hundreds of electrical items, such as motor starters, switchgear, transformers, auxiliary generators, etc., scattered throughout the plant.
- Main Condenser Heat Rejection System: Reactor salt coolant system, fuel-salt system, intermediate coolant system, power system, reheaters, coolant supply and treatment were projected here.

Table 2. Economic parameters of building a 1000 MW(e) MSBR ^[25, 26].

	1978			2000			2011		
	MSBR	PWR	Coal	MSBR	PWR	Coal	MSBR	PWR	Coal
Direct Costs									
Land and Land Rights	1	2	2	3	5	5	4	7	7
Structure & Improvements	29	111	245	77	293	647	102	391	863
Reactor Plant Equipment	90	139	–	238	367	–	317	489	–
Turbine Plant Equipment	45	113	88	119	298	232	158	398	310
Electric Plant Equipment	6	44	31	16	116	82	21	155	109
Miscellaneous Plant Equipment	2	13	11	5	34	29	7	46	39
Main Condenser Heat Rejection System	–	22	14	–	58	37	–	77	49
Total Direct Costs	173	444	391	457	1173	1033	609	1563	1377
Indirect Costs									
Construction Services	–	70	39	–	185	103	–	247	137
Home-Office Engineering Services	–	53	16	–	140	42	–	187	56
Field-Office Engineering Services	–	30	10	–	79	26	–	106	35
Total Indirect Costs	67	153	65	177	404	172	236	539	229
Total Capital Cost	240	597	456	634	1577	1204	845	2102	1606

7.2. Cost of Electricity

Table 3 displays the cost of electricity with inflation for a MSBR based on preliminary estimates at ORNL and compared to their PWR and coal plant estimates of the same pre-1980 vintage plants. The results were 5.48, 5.86, and 5.96 ¢/kWh for MSBR, PWR, and coal, respectively, and values were obtained from Equation 1 below^[27, 28].

$$COE = \frac{(C * i + fuel + O \& M + WD + DC)}{P_e * 8760 * C_f} \quad (1)$$

Where COE is the cost of electricity in \$/MWh or ¢/kWh, C is the capital cost in dollars, i is the fixed capital charge rate (typically 10%), $fuel$ is the annual cost for fuel, $O\&M$ is the annualized operations and maintenance cost, WD is the waste disposal cost, DC is the decommissioning cost, P_e is the net electrical power capacity of the plant, and C_f is the capacity factor. The capacity factor is taken as 0.9 for the MSBR to account for the reduced down-time because of its on-line fueling feature and 0.8 for both PWR and coal. There are 8760 hours per year. To convert costs quoted in one year's dollars to those in another year we used deflation factors. For example, for 1978 dollars converted to 2011 dollars, one could use the multiplier of 3.45^[26]. If the capacity factor for the MSBR increases from 0.9 to 0.95 its COE becomes 3.65 ¢/kWh^[28]. The net plant efficiency was assumed to be 44% for MSBR and coal while 33% efficiency was assumed for the PWR^[27, 28]. Early designs of the MSBR proposed using a steam cycle for electricity production but current proposals for an MSBR favor a Brayton cycle^[29].

Table 3. Cost of electricity production ^[26-28].

	¢/kWh								
	1978			2000			2011		
	MSBR	PWR	Coal	MSBR	PWR	Coal	MSBR	PWR	Coal
Capital	0.83	0.85	0.65	2.19	2.24	1.72	2.86	2.93	2.24
Operation and Maintenance	0.24	0.47	0.33	0.63	1.24	0.87	0.83	1.62	1.14
Fuel	0.46	0.31	0.71	1.21	0.82	1.87	1.59	1.07	2.45
Waste Disposal	0.04	0.04	0.04	0.11	0.11	0.11	0.14	0.14	0.14
Decommissioning	0.02	0.03		0.05	0.08		0.07	0.10	
Total	1.58	1.69	1.73	4.20	4.49	4.56	5.48	5.86	5.96

7.3. Licensing

Although two experimental MSBRs have been built and operated in the United States under government ownership, none have ever been subjected to formal licensing or even detailed review by the Nuclear Regulatory Commission (NRC). As a consequence, the question of licensability of MSBRs remains open; the NRC have not yet identified the major licensing issues and the concept has not been considered by various public interest organizations that are often involved in nuclear plant licensing procedures. Presumably, MSBRs would be required to comply with the intents, rather than the letter, of NRC requirements, particularly where methods of compliance are concept-specific ^[25].

8. Proliferation-Reducing Aspects of Thorium-Based Fuels

The students identified that one of the main proponents of thorium-based fuels was the proliferation-reducing aspects of the fuel as well as the safety and benefits discussed in the next section. This portion of the curriculum would have been supplemented with a paneled debate and role-playing.

With the continuing concern around nuclear weapons proliferation, any new technology must be developed with attention to potential proliferation risks. Currently, the United States is engaged with the International Atomic Energy Agency (IAEA) and allies around the world to control the spread of highly-enriched uranium (HEU), which is defined as uranium with ^{235}U content higher than 20% ^[6]. In addition to the reduction in proliferation risks of fuel production, the contents of the spent fuel should be considered. Of particular concern in the uranium fuel cycle is the disposal and security of the spent fuel. The solid state of uranium fuel makes it particularly susceptible towards potential use by a terrorist organization as material for a “dirty bomb.” Additionally, depending on how uranium fuel is used in a reactor, there is the potential for significant plutonium production, which could lead to the construction of nuclear bombs. The thorium fuel cycle has three main proliferation reducing points, namely that the ^{233}U in a MSBR would be burned in-situ, ^{232}U would be inevitably produced and the quantity and quality of plutonium production is reduced.

Thorium-based fuels are not only “self-protecting” but utilization of such fuels reduces the production of spent fuel waste up to 50% and simultaneously reduces the amount of plutonium buildup in the fuel ^[30]. The primary mode of proliferation resistance for thorium fuel comes from the high radioactivity of the spent fuel, which is said to be self-protecting. One main security feature of the fuel is that 36% of ^{232}U decays will have a very energetic gamma that makes manipulation of the fuel salt and uranium very difficult even in small amounts ^[31]. After use in a reactor, the fuel will contain the main fertile species ^{232}U , which decays to ^{208}Tl , accounting for 85% of the total dose from ^{232}U after two years ^[30, 32]. In comparison, weapons-grade plutonium containing about 0.36% ^{241}Pu (decaying to ^{241}Am emitting <0.1 MeV gamma) does not present a significant radiological hazard although reactor-grade plutonium having about 10% ^{241}Pu does. The ^{208}Tl daughter is a high-energy gamma-emitter of about 2.6 MeV, which requires extensive shielding and remote handling. The high radioactivity means that the fuel is not only self-protecting, but also easy to detect even in small quantities should it need to be tracked down, which makes it less desirable than plutonium as weapons material ^[30, 32]. Additionally, if the fuel is contaminated with ^{232}U , the ^{233}U in the fuel will not make a suitable weapons material. From a reactor design perspective, there is no motivation to separate the two

isotopes to have a successful power plant ^[32]. Table 4 below compares the unshielded working hours required to accumulate a 5 rem dose from plutonium versus ²³³U and ²³²U sources ^[30]. In the same vein, Table 5 displays the thickness of lead required to shield 1 kg of ²³³U ^[6].

Table 4. Unshielded working hours required to accumulate a 5 rem dose from 5 kg sphere at 0.5 m a year after separation ^[30].

Metal	Dose rate (rem/h)	H
Weapons-grade plutonium	0.0013	3800
Reactor-grade plutonium	0.0082	610
²³³ U containing 1 wppm ²³² U	0.013	380
²³³ U containing 5 wppm ²³² U	0.059	80
²³³ U containing 100 wppm ²³² U	1.27	4
²³³ U containing 1% ²³² U	1.27	0.04

Table 5. Thickness (cm) of lead needed to shield 1 kg of ²³³U as a function of the ageing time (d) after chemical separation ^[6].

²³² U/ ²³³ U	10 d	30 d	100 d	400 d	4000 d
100 ppm	6.75	10.4	13.3	16.4	18.4
1000 ppm	12.1	15.5	18.4	21.5	23.5
10000 ppm	17.2	20.8	23.5	26.0	28.1

Perhaps the strongest proliferation-reducing feature for thorium fuel for the MSBR is the fact that the fuel salt is in liquid form. Fuel in a liquid form would be highly difficult for an organization to steal without significant planning and preparation. Even then, the need to keep the material in a liquid form makes it nearly impossible to use for any purpose other than in nuclear reactors. Together, these factors make thorium based liquid salt reactors far more proliferation resistant than their current uranium counterparts ^[33].

The thorium fuel designs are regarded as a superior alternative for nuclear power production in terms of the plutonium proliferation risk that is inherent in the spent fuel of uranium-based reactor designs. A thorium-based reactor would average about 92 kg of plutonium per GW-yr of electricity generated versus a conventional light water reactor that would result in 232 kg of plutonium per GW-yr of electricity generated. The spent fuel of thorium-based reactors would contain a lower percentage of ²³⁹Pu and higher percentages of undesirable ²³⁸Pu, ²⁴⁰Pu and ²⁴²Pu, making spontaneous fission more likely. The IAEA defines weapons-grade plutonium as requiring 90% ²³⁹Pu ^[6]. Thus, a thorium process is less suitable for development of plutonium-based weapons. If developed, the MSBR could prove resistant to diversion of special nuclear materials in direct comparison to that of solid-fueled reactors without fuel reprocessing ^[34]. The MSR probably does not offer significant advantages over other reactor systems as a single, stand-alone diversion-proof reactor to warrant development for that purpose only. However, in a broader sense in terms of diversion of plutonium from US reactors, the MSBR offers an attractive option to limit the total plutonium buildup. The end

products of plutonium charged in a MSBR would be essentially free of ^{239}Pu , ^{240}Pu and ^{241}Pu with about 12% of the original mass remaining as ^{242}Pu or higher-mass capture products ^[34].

9. Safety and Benefits

The primary safety concern facing today's uranium based reactors is the potential for fuel melt-down. A melt-down refers to a situation where the cooling capacity to the reactor core has been disrupted, and excess heat can no longer be removed from the solid fuel plates. If left unchecked for a long enough period of time, the solid uranium fuel will eventually become molten, and melt through the reactor pressure vessel, releasing dangerous radioactive isotopes into the environment. After the Fukushima Daiichi accident in 2011, we learned that a small partial melt-down of the fuel probably caused hydrogen to be formed from the interaction of the fuel's very hot zirconium cladding with steam after the fuel became uncovered ^[35]. The great benefit of a MSBR is that a melt-down in the traditional sense cannot happen. With the fuel already in a molten state, the entire reactor system is designed to handle the fuel at these high temperatures ^[32]. Additionally, unlike a traditional uranium based reactor, should an MSBR shut down, the fuel will cool and eventually solidify. Then, the fuel can be re-melted and pumped back into the reactor.

Another downside of many of today's reactors is the high pressure found in the coolant loops. These high pressure systems are necessary to avoid steam generation in the coolant loop. Should pressure be lost in the coolant loop of a uranium reactor, there would be a great risk of coolant boiling at the surface of the fuel, thereby significantly reducing the heat transfer from the fuel to the coolant. In a MSBR, because the molten salt is used as the primary heat transfer fluid, there is no need to pressurize the system, as it will be operating well below the boiling point of the salt. Therefore, there is no risk of the type of accident scenario seen in a PWR loss of pressure accident ^[36]. Additionally, because the molten salt loop is not pressurized, should there be an accident involving a burst pipe in the reactor, there is no risk of the fuel being pushed out of the reactor system from a high pressure force. If a burst pipe is detected in the reactor, the system can be designed to immediately shut down the fuel circulation pumps, thereby controlling any further release of molten fuel from the fuel loop.

10. Challenges

One of the greatest drawbacks in the use of thorium fuel is the need to constantly reprocess the fuel and at high volumes. If undesirable species were left to decay in the fuel, they would inhibit the ability to produce ^{233}U in the quantities necessary for use in power production. This problem could be solved through development of continuous and economic chemical processing of the fuel. Since the fuel is in a liquid form in the reactor, it is theoretically possible to cycle the molten salt fuel through some mechanism to reprocess the fuel while the reactor is still operating. This process would have the potential to produce nearly pure ^{233}U , and avoid the issues surrounding protactinium buildup. However, while it is theoretically possible to have on-line processing of the fuel, the economics and science are still unproven and require further study. Another important factor for consideration is the time it takes to breed enough ^{233}U to fuel a new core—the reactor doubling time is about 20 years.

As with any nuclear reactor, it is vitally important to remove excess heat from the reactor system. In the case of a MSR, the liquid fuel is used as the heat transfer medium, with the liquid

fuel being pumped from the reactor core to a nearby heat exchanger. In PWRs, because the fuel is maintained in a solid form, water is used as the coolant, and is piped over the fuel in the reactor core before going to an outside heat exchanger. This setup has the advantage of not having any radioactive material in the active coolant loop. In the case of an MSBR, a heat exchanger will need to be located in the containment building to transfer heat through a heat exchanger to some intermediary fluid for heat transfer to outside the containment building. This two-step process is relatively well understood, and should be simple to address. The most critical factor will be to ensure that the heat exchanger in the containment structure is made of materials which are robust enough to withstand the high radiation levels present from the hot fuel. Although, the molten fuel salt itself has shown negligible radiolytic damage, studies on the radiation performance of superalloys and nuclear graphite may show otherwise. Therefore, studies involving the projection of service life for MSBR components are necessary.

In terms of disaster preparedness, additional work will be necessary to analyze specific accident scenarios. Due to the liquid nature of the fuel, it is conceived that in a loss of power or loss of coolant event, the molten fuel could be gravity fed into underground storage tanks. These tanks will need to be designed to prevent the fuel from solidifying in a shape which could be naturally critical. The final report from the Oak Ridge MSRE mentions but makes no specific recommendations regarding this issue. Additionally, it is necessary to address the implications of a structural failure (i.e. pipe burst) within the containment building. A burst pipe or a cracked heat exchanger would result in loose molten fuel that is highly radioactive in the containment building. Due to the high radioactivity of the fuel, it would be difficult to clean up the accident. Therefore, steps must be taken to ensure that the physical design of the containment structure can appropriately handle the containment and control loose fuel.

Because of the inability for humans to enter the containment area once the reactor is started, it will be necessary to provide remote handling and operation of all systems within the containment building. This is made especially difficult due to the potential for electronics to fail in a highly radioactive environment. Therefore, more work is necessary to address the options for autonomous operation of reactor systems in a high radiation environment. Additionally, since anything that goes into the containment building will likely never come out, it will be vitally important to ensure that in-situ storage capacity is built into the reactor design for any parts that may fail or need to be replaced during the lifetime of the reactor.

This project was a multifaceted assignment for a course on advanced nuclear fuel cycles at a graduate level. The research project immersed students in local history, technical reactor research, voluminous journal articles, intensive writing and economics. A primer was created from this project and it was co-written by the students. The final stage of the project was outreach and presentations. The students organized their meeting notes and created a final brochure to highlight main points of the primer.

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