Increasing the Mobility of a Small Modular Molten Salt Nuclear Battery

A Thesis
Presented in Partial Fulfilment of the Requirements for the
Degree of Master of Science
with a
Major in Nuclear Engineering
in the
College of Graduate Studies
University of Idaho
by
David H. Haar

Approved by:
Major Professor: Richard Christensen, Ph.D.
Committee Members: Indrajit Charit, Ph.D.; Haiyan Zhao, Ph.D.
Department Administrator: Indrajit Charit, Ph.D.

December 2021

Abstract

The concept of Small Modular Mobile Reactors (SMMR) has been around since the beginning of the atomic age. In remote locations, electrical power and heat provided by conventional power systems require constant resupply of bulky fuel such as coal or diesel. These shipments often constitute the major expense of keeping a remote location operational. A small nuclear reactor eliminates the need for these shipments.

However, the intense radiological source term at the end of life associated with all nuclear reactors makes mobility challenging, and makes decontamination and decommissioning very expensive, difficult, and time consuming. As a result, the deployment of these reactors has been very limited, and many of the experimental SMMRs constructed in the 1950s and 1960s have yet to be decommissioned.

For SMMRs to be practical, they must be capable of being moved relatively easily and quickly, and when they reach the end of use, they must be able to be disposed of quickly.

This thesis will present a transportation and disposal analysis of an SMMR referred to as the Molten Salt Nuclear Battery (MSnB), based on natural circulation of a molten fluoride salt (FLiNaK), operating at 400 kW thermal for a period of 10 years. Further, it will present the design changes necessary to allow the reactor to be recovered from the deployment location within weeks of reactor shutdown, using only over-the-road and currently available transportation equipment, transport the spent fuel to a location for either reprocessing or long term storage, and dispose of all other reactor components immediately as low level waste.

Acknowledgements

I would like to thank my family for their constant support, encouragement, and patience through development of this thesis.

Further, I want to thank Dr. Richard Christensen for sharing his enthusiasm for this project as well as providing direction and clarity, and to Alice Allen for her endless patience in guiding me through the graduate school experience.

Table of Contents

List of Tables

List of Figures

CHAPTER 1

Introduction

1.1 Problem Statement

A small portable reactor, capable of rapid deployment and recovery to remote locations, has been a goal within the nuclear community since the beginning of the nuclear age. Such a reactor could be used to provide energy to isolated areas, such as remote military installations, sparcely populated areas without significant infrastructure, areas ravaged by natural disasters such as earthquakes, floods or hurricanes, and underdeveloped areas without access to clean water or electrical power. The energy provided could be used to desalinate water, provide heat, or provide electrical power.

While numerous small reactors have been successfully built and operated, none have had any success with rapid mobility and disposal. This is due entirely to the radiological source term inherent to any nuclear reactor, composed of fission products, activation products, and actinides. Both the fission and activation products produce radiation fields in excess of 10,000 REM/hour for a typical reactor, far beyond lethal thresholds, necessitating massive shielding for personnel and the need to use remotely operated equipment. When the system is breached (i.e., opened to the atmosphere), radiological contamination becomes an issue, necessitating containment, HEPA filtered ventilation, personnel protective suits and respiratory equipment, and monitoring systems. The personnel performing this work must be skilled and knowledgeable, and the management must be technically competent. Numerous regulations and requirements govern this type of work, and these protections are not limited to the personnel performing work. The general public is also potentially exposed during off-normal events during the decontamination and decommissioning operations, as well as during transport and disposal operations.

In order to provide the necessary controls to ensure compliance, the organization performing the work must have a large and diverse technically competent staff populating several departments, including engineering, training, emergency management, emergency response, criticality safety, nuclear safety, legal, scheduling, budgeting, industrial hygiene, occupational safety, medical, radiological controls, environmental monitoring and reporting, administrative, procurement, information management, and quality assurance. As a direct result, decontamination and decommissioning activities become complicated, time-consuming, and expensive. It is not uncommon for the decommissioning activities to cost as much or more than the original reactor construction, and for decades to pass before decommissioning is completed.

1.2 A Brief History of Small Modular Reactors

In the period following World War II, the United States initiated several efforts to develop nuclear systems for power generation rather than weapons [1]. Most of these were military in nature. The air force initiated development of a nuclear powered plane, the navy initiated development of a nuclear powered submarine, and the army initiated development of several small reactors to provide a range of needs, from a truck mounted reactor capable of being rapidly deployed and recovered within a combat theater environment to a reactor capable of providing power and heat to a remote military base. Argonne National Laboratory was established during this time, and given the task to develop civilian power reactors, although none of these were designed for mobility.

The air force's Aircraft Nuclear Propulsion (ANP) program operated for 14 years, from 1949 until 1963 [2]. While much development work was accomplished, the actual aircraft was never constructed. The program was terminated for two primary reasons; the first was excessive weight due to the shielding necessary to protect the flight crew during operations, and the second was the advancements in intercontinental missile system capabilities, which were deemed superior to airplanes. In addition, the concern about a crash over land resulting in the wide dispersal of nuclear materials, the exposure of many people, and the impossibility of clean up was never satisfactorily addressed. As such, the inability to safely address mobility was another significant reason for the termination of this program.

The naval program was very successful, and is still operating today. The obvious tactical advantages, the shielding provided by the surrounding water, and the lack of severe weight limits all contributed to solving the technical issues of incorporating a small reactor into a ship design. Perhaps most importantly from a mobility perspective, the robust design requirements necessary for a submarine to survive ocean pressures, the relative low speeds of operation, and the reactor pressure vessels used in pressurized water reactors allowed the reactor to survive accident scenarios. The US Navy has lost two nuclear submarines; the Thresher and the Scorpion. The Thresher was lost during sea trials, and suffered a hull collapse. The Scorpion was lost during operations, and is assumed to have either collided with an underwater mountain or suffered a torpedo accident. In both cases, analytical studies have concluded the reactor pressure vessel was not breached, and fell to the ocean floor intact. No elevated radiation levels have been observed in the areas where either sub was lost [3].

The Army Nuclear Power Program (ANPP) operated from 1954 until 1976 [4]. It constructed, tested and operated eight reactors with a wide variety of modularity and mobility.

The first was the Stationary Medium Power Reactor Number 1 (SM-1) constructed in Fort Belvoir, Virginia, at a cost of approximately 2 million dollars. This PWR plant was the prototype for reactors capable of providing power and heat to an entire base. It produced 10 MWt and 1.85 MWe. It went critical in 1957, and operated until 1973, when it was defueled. Decommissioning was delayed on the assumption that as the shorter lived isotopes continued to decay, future decommissioning activities would produce less waste and result in lower amounts of radiological exposure to personnel. The plant still stands today. In 2020, the U.S. Army Corps of Engineers awarded a 68 million dollar contract for the decommissioning, dismantling and disposal of the deactivated SM-1 nuclear power plant. Total removal is projected to be completed by 2025 [5].

Next came the Stationary Low Power Reactor Number 1 (SL-1). This was a smaller plant intended for use at the remote radar stations near the arctic circle as part of the Distant Early Warning (DEW) system. Because size and weight were of paramount importance, this plant was designed as a BWR. It was designed by Argonne, and constructed at the National Reactor Testing Station in Idaho. It operated from 1958 through 1960. In early 1961, it suffered a steam explosion when the center control rod was withdrawn too far. Because the reactor was destroyed and the core opened to the atmosphere, is was completely dismantled during 1961 and 1962, and most components were disposed in a specifically created landfill on site [2]. The fuel modules were transferred to the Idaho Chemical Processing Plant. This accident effectively terminated the Army program, though not immediately. There were several other reactors in various stages of development, and although several were far enough along that they were actually constructed and operated, most of the plants in the early design phase were terminated within a couple of years.

Stationary Medium Power Reactor Number 1A (SM-1A) was a 20.2 MWt, 1.64 MWe PWR which provided heat and electrical power at Fort Greely, Alaska. It was based on the SM-1, modified to produce more heat. It operated from 1962 until 1972, when it was defueled. Like SM-1, decommissioning was delayed to save waste and exposure. It still stands today. Decommissioning is supposed to begin in 2022 and is estimated to require 6 years [6]. As it is larger than SM-1, and in a much more difficult work location, costs are expected to be at least as much as SM-1.

Portable Medium Power Reactor 2A (PM-2A) was a 1.5 MWe PWR designed to provide electrical power (including electric heat) to Camp Century in Greenland [7]. It was designed to be installed below grade within an ice tunnel. This was the first time reactor design focused on modular construction to minimize the assembly time at the deployment location, and make transportation easier. Accordingly, each module was sized such that any mode of transportation could be used; air, land, or sea, using lifting and handling equipment subject to the same transport limitations. With the exception of the core, the plant was fully assembled and tested in the States, component handling and assembly procedures were developed, and packaging configurations designed. 27 separate modules were required for plant assembly. All but one of these was transported via ship to Greenland. The heaviest module was shipped via military cargo aircraft to prove the viability of air transport. Upon arrival at Greenland, all modules were pulled across the ice cap via trailers and sleds.

Total procurement and assembly costs were 5.7 million dollars. The reactor became operational in February of 1961. The ice movement was greater than expected, and year-round operations at Camp Century ceased in 1963, eliminating the need for the reactor plant. During the summer of 1964, the nuclear plant was disassembled and shipped to the United States. According to a technical report issued by the Army Materiel Command in 1965 [8], "...residual radiation levels around the primary unit (i.e., reactor and hot waste tank) were considerably higher than had been anticipated. Hence, daily permissible exposure of crew members disassembling these components was shorter than had been calculated, and, as a result, more personnel were required to accomplish the task in time to meet scheduled shipping dates than was planned originally." The radiological exposure limits at the time were 5 REM per year of whole body exposure. From this statement, I assume many personnel received the full 5 REM.

The treaty between the Danish and US governments severely restricted the amount of radioactive waste which could be released at the camp. All solid waste was prohibited, and liquid waste was limited to 50 millicuries per year. The reactor systems were shipped to the National Reactor Testing Station. The primary skid went to the Test Area North for destructive evaluation, and all other radioactive materials were shipped to the burial grounds.

Seasonal operations continued until 1967, when the base was abandoned with minimal decommissioning, as engineering design of the era assumed that the base would be preserved for eternity by perpetual snowfall. Because of climate change, the base is now being exposed, and discussions are occurring between the US and Denmark concerning responsibility for the cleanup, and the extent of cleanup required.

The Portable Medium Power Reactor number 3A (PM-3A) was a 1.8 MWe PWR which provided electrical power to the McMurdo Sound Experimental Station in Antarctica, as well as steam for the desalination of 14,000 gallons of drinking water per day. It was also designed as a modular unit, capable of being transported via military aircraft, although I do not believe it ever was transported in this manner. 33 separate packages were required to deliver the system. It became operational in 1962, required 2 years to fully complete testing and debugging, and operated until 1972. It suffered 438 malfunctions during the 8-year operating period, and was permanently shut down due to corrosion issues. As with Greenland, an international treaty prevented disposal of radioactive materials in Antarctica, so the material was shipped back to the states. The corrosion issues had resulted in the leakage of primary coolant, contaminating the rock used to shield the reactor, as well as the surrounding soil. Cleanup required 5 years to complete due to the limitation of summer work only, and 365 metric tons of radioactive waste were shipped to the States. This number included 9,000 cubic meters of significantly contaminated rock sent to the Savannah River Site, and 14,000 cubic meters of low level contaminated soil to a base in California [9].

Portable Medium Power Reactor 1 (PM-1) was a 1.0 MWe, 10 MWt PWR which provided power and heat to a remote radar installation near Sundance, Wyoming [1]. Another modularized designed for military air transport, it consisted of 16 separate containers, all of which were transported by air using C-130 planes to Rapid City, SD, and then transported via truck to the top of Warren Peak. To save weight, the reactor was designed to be installed below grade, such that the surrounding earth would provide the necessary shielding. It operated from 1961 until 1968, when it was shut down for budgetary reasons. The plant appears to have operated without incident. The spent fuel was shipped to the SRS for reprocessing. I found no information on the decommissioning activities.

The Mobile Low Power Reactor 1 (ML-1) was a 0.33 MW closed-cycle gas turbine tested at National Reactor Testing Station from 1961 until 1965 [2] [10]. This new and innovative design was selected to eliminate the heavy components of a PWR. This reactor was composed of 4 major skid mounted components: the reactor, the power conversion unit, an instrumentation and control unit, and auxillary equipment. It was designed to be deployed using no more than 6 skids, capable of being set up at the deployment location in less than 12 hours, and ready for recovery and relocation in less than 6. While the program suffered many problems, nearly all were associated with the pressurized gas turbine components, and were not unusual for a first of a kind system. The most significant issue from a mobility standpoint was the radiological dose rate to personnel, both for operation and transport. A 15-ton weight limit was imposed on the reactor skid for transport. The reactor shielding consisted of a 2-inch thick lead blanket to serve as a fast neutron reflector, then a 3-inch thick layer of borated water to act as the primary neutron shield, then another 2-inches of lead, all encased in a 9-foot diameter tank filled with water. The water was drained prior to transport to comply with the 15 ton weight limit. Additional tungsten sheilding was used to protect the driver during transport, such that the dose rate in the cab was limited to 15 mr/hr within 24 hours of reactor shutdown and water removal. Dose rates in the opposite direction were approximately 10 times higher. The reactor was designed to be remotely operated by a control station located 500 feet from the reactor, such that the dose rate in the operations center would be approximately 30 mr/hr when the reactor was at full power. Assuming the outer surface of the shield was 6 feet from the reactor center, the dose rate at contact with the outer

Figure 1.1: ML-1 Reactor Skid

shield would be

$$D_{shield} = D_{controlcenter} d_{cc}^2 / d_s^2 = 30(500^2)/6^2 = 2.08(10)^5 mr/hr = 208R/hr$$

The only mention the design report makes of the space between the reactor and the operations center is that "...it may be desirable to employ some expediant shielding when operating the power plant in the field." [10].

The final reactor developed under the ANPP was the Mobile High Power Reactor 1A (MH-1A). This was a barge mounted 10 MW PWR intended to provide electrical power along coastal areas. It cost about 17 million to construct, and was used from 1968 until 1976 to provide electrical power to the Panama Canal zone, where drought conditions had significantly reduced available hydroelectric power. It was towed back to Ft. Belvoir in 1977 and defueled, sending the spent fuel to SRS for reprocessing. The plant was put into safe storage until 2014, when a 35 million dollar contract was awarded to decommission the barge. The work was completed in 2019.

1.3 Recent Developments

At least within the US, the development of mobile modular reactors has been dormant for many years. However, in 2020, the Department of Defense awarded a total of 40 million dollars to three different vendors to once again develop a mobile modular reactor for deployment to remote bases as part of Project Pele. While the objectives for the original program from the 1950s remain unchanged, an additional need has been identified in the space program for power needs on planets other than earth, specifically Mars.

The March 9, 2020 press release associated with the contract awards by the DOD states in part:

"The Pele Program's uniqueness lies in the reactor's mobility and safety" said Dr. Jeff Waksman, Project Pele program manager. "We will leverage our industry partners to develop a system that can be safely and rapidly moved by road, rail, sea or air and for quick set up and shut down, with a design which is inherently safe."

In October of 2018, the US Army issued a document entitled 'Study on the use of Mobile Nuclear Power Plants for Ground Operations' [11]. There is no discussion in this document concerning waste management, other than a brief statement about which government has the responsibility for disposal if the reactor is deployed outside of US territory. It would appear that the difference between this latest effort and the previous effort is limited to the inherent safety advances in the generation IV reactor plant designs, such as the use of TRISO fuels. Air transport is still clearly viewed as feasible, and there is nothing to indicate waste is a significant design criteria. From my perspective, this has been a crippling limitation to the nuclear industry in the United States. Nuclear plant designs focus on reliability and performance, and ignore what will be done with the reactor components at end of life. As with most waste management issues across the globe in nearly all industries, the waste is left as the next generation's problem to solve.

Of all the lessons from the experience gained in the 1950s and 1960s, I think several are of vital importance:

(1) Delaying waste disposal is very nearly always a mistake for several reasons: disposal regulations tend to become more restrictive with time rather than less; allowing the source term to decay does not significantly reduce the level of radiological controls required to perform work; and institutional knowledge may be lost which can complicate disposal. Accordingly, a plan for disposing of all reactor materials must be developed as part of the reactor design.

(2) Field assembly in remote or inhospitable locations is difficult and time consuming, and recovery from accident scenarios is even more so. As such, the reactor must consist of the fewest number of components practical, be very robust, and require minimal equipment for assembly. Further, the assembly and disassembly should focus on 'plug and play' type designs, meaning the components fit together without intrusive operations

like welding.

(3) Decontamination and decommissioning is time consuming, expensive and difficult. It must be incorporated into the reactor design from the very beginning. This requires that modular components can be separated from each other without the need for radiological contamination control systems, and remote operations be used in high radiation environments.

(4) Transportation and disposal activities should comply with current regulatory requirements without relying on special exemptions. This effectively eliminates the use of air transport.

(5) For a reactor to be truly mobile, it should be able to be deployed using standard commercially available over-the-road transportation equipment, and should not require significant infrastructure to unload from the transport vehicle and assemble.

The remainder of this thesis will apply these lessons to the design of the MSnB being developed by the University of Idaho, and show that it is possible to develop a reactor which can comply with all of them.

CHAPTER 2

Regulatory Framework

Two primary sets of regulations are relevant to the transportation and disposal of reactor components.

2.1 Transportation

Transportation regulations are designed to protect the environment, public, transportation work force personnel, and any emergency responders in the event of an accident. The primary risk is radiological contamination, as it presents internal pathways for radiological materials from both inhalation and ingestion, making the time of exposure much greater and exposing much more sensitive biological tissues. Further, contamination is widely dispersed and difficult if not impossible to control once released. The secondary risk is whole body dose, and is only secondary because the sources tend to be discrete and fixed in place, making it easier to control exposure.

Packaging and transport of radioactive materials is governed by several different sections of the Code of Federal Regulations (CFR). There are two that are of primary importance to the MSnB. The first is 49 CFR, parts 171 through 178, which detail the Department of Transportation (DOT) requirements for any hazardous material shipment. Radioactive materials are a subset of hazardous materials, and are specifically covered under 49 CFR 173, subpart I. These regulations specify how hazardous materials will be marked, packaged, labeled, manifested, etc.

In general, as the potential radiological exposure to personnel from various accident scenarios increase, the risk is mitigated by placing the radioactive material inside increasingly robust transport packages. The packages are designed to ensure that any release of radiological material is within acceptable limits. The regulations define several packaging levels, starting with no packaging (e.g., dump truck) for naturally occurring radioactive material, such as ores, exempted packaging for limited quantities of radioactive material, such as residential smoke detectors, Industrial Packages (IP) for Surface Contaminated Objects (SCO) or Low Specific Activity (LSA) components, Type A packages for higher activity shipments, fissile material packages for unirradiated special nuclear material, and Type B packages for the highest activity shipments. Packages at the low end of this scale are not expected to survive accidents intact, while packages at the high end are required to survive all plausible accidents without release of any radioactive material.

The second set of radioactive material transport regulations is provided in 10 CFR, part 71, which details the Nuclear Regulatory Commission (NRC) requirements for the testing, certification and use of the higher activity transport packages (i.e., LSA category III, Type A, Type B, and fissile material packages). The NRC has developed two specific activity limits for all radioisotopes, referred to as A1 and A2 values, which define the maximum amount of activity that is allowed to be transported in a Type A container. A1 applies to what are called Special Form materials, where the radiological source material is clad with a robust protective layer capable of withstanding the applicable accident scenarios (e.g., sealed radiological sources used for instrument calibration). A2 applies to the more general case of isotopes which could become mobile during the applicable accident scenarios. The LSA category III (solids) is intended for activated components, where the radiological source material is immobilized and dispersed throughout a larger volume of non-radioactive material, as is the case with activated components. The LSA-III specific activity limits are 0.002 times the A2 value per gram of material. Quantities that exceed the A1 or A2 values require the more robust Type B package.

The NRC has defined a series of tests which bound all plausible accident scenarios for Type B containers. The three most severe accidents are: (1) the maximally loaded package is dropped from a height of 30 feet onto a non-yielding surface such that the most vulnerable portion of the package impacts the surface; (2) the entire package is subjected to at least 1475 degrees F for a minimum of 30 minutes; and (3) the entire package is submerged under at least 50 feet of water for a minimum of 8 hours. All Type B packages are tightly sealed to isolate the contents from the environment during transport.

Most Type B packages are designed around a specific payload, because the larger the weight, the more difficult it is to survive the tests intact. If a package is built for a more general purpose, the empty space between the payload and the interior package walls must be filled with shock absorbing materials (internal impact limiters) to prevent dynamic loading which could exceed the bounding tests. The weight of the impact limiters becomes part of the allowed payload, and may necessitate further testing if modeling and analysis used to demonstrate the package still survives the accident scenarios are inconclusive.

Air shipment of radioactive materials is not regulated by the Code of Federal Regulations, as are most other methods. Regulations for air transport are issued by the International Atomic Energy Agency (IAEA). Air transport is severely limited to small well-packaged materials where either the accident consequences are deemed to be insignificant or a Type C shipping package is used. Type C packages are even more robust than Type B packages, as they must survive being dropped from an aircraft at cruising altitude without losing integrity. In the US, air shipments of radioactive material consist primarily of small quantities of radiopharmaceuticals. There are specific exemptions for military use such as nuclear weapons transport. It is for this reason that I think any type of air transport for the MSnB is unrealistic. Such transport would only be allowed by special exemption, and there is no good rationale for such an exemption that will be readily accepted by the general public.

Water transport is also regulated by the IAEA, with support from the International Maritime Organization. In general, this requires purpose-built ships for spent fuel transport using at least Type B packages.

The primary factors that drive the selection of transportation equipment for the reactor are therefore: (1) weight; (2) radiological dose; (3) mobility of the source term; (4) physical size; (5) decay heat; and (6) quantity of fissile material.

2.2 Disposal

Radioactive waste disposal regulations are intended to isolate the source term from the biosphere until it has decayed sufficiently to no longer pose a concern to either personnel or the environment. It is therefore based on the time required to decay, the mobility of the various isotopes once released into the environment, and the resistance to degradation of the waste form. Radioactive material generation, use, ownership, and disposal responsibilities are primarily regulated by the Atomic Energy Act of 1954. The atomic energy commission was eliminated by the Energy Reorganization Act of 1974, which replaced it with the NRC and the Department of Energy (DOE). This act assigned the responsibility for nuclear weapons development and the promotion of nuclear energy production to the DOE, and assigned responsibility for establishing regulatory requirements and licensing activities to the NRC. The NRC is therefore responsible for establishing the waste disposal regulations.

The NRC defines the specific categories of radioactive waste as follows:

2.2.1 High Level Waste

High Level Waste (HLW) consists of spent nuclear fuel as well as the byproducts of reprocessing spent fuel. This material is required to be isolated from the biosphere for a very long time (current EPA standards require dose determinations for 1 million years), so the waste form must be extremely stable; i.e., resistant to corrosion and other forms of decomposition, and unlikely to allow migration of radioactive materials from the disposal location. This requirement is due almost entirely to the Actinides, as they tend to have very long half-lives and complex decay chains, but there are some fission and activation products with long half-lives and high mobility in the environment as well, such as Tc-99, Ni-59 and Ca-41.

The Nuclear Waste Policy Act (NWPA) of 1982 established a national program to develop a necessary repository for HLW. The act assigned DOE the responsibility for siting, operation and closure, the EPA the responsibility to develop performance standards, and the NRC the responsibility for licensing of the repository. The 1987 amendment to the NWPA reduced the nine sites under study to the single site of Yucca Mountain in Nevada. Although all three of the involved agencies have completed the bulk of their responsibilities under the NWPA (including the determination by the NRC that Yucca as designed complies with the waste isolation requirements), political opposition to this decision has halted further action to date. As such, no licensed repository exists in the US, and all such material remains primarily at the sites of generation.

Nonetheless, whether Yucca Mountain eventually comes to pass or a different HLW repository is established, the manner in which the waste will be isolated from the biosphere will probably be very similar. Two criteria drive the design of all HLW containers. The first is the waste form. Due to the required stability of the waste form, the draft Yucca Waste Acceptance Criteria (WAC) requires borosilicate glass for HLW and corrosion resistant metal clad fuel assemblies for SNF. The second is radiological dose. Due to the extreme dose rate associated with HLW, a series of nested canisters are used to eliminate the need to transfer waste from one canister to another. First, the stabilized waste material is placed inside a shielded canister. The shielded canister is then placed inside another canister for transport, and a yet a different series of canister(s) for emplacement within the disposal repository. Due to the wide variety of HLW/SNF configurations already in existence, a Multi-Purpose Canister (MPC) is under development to help standardize and simplify HLW handling and disposal.

As such, there is currently no disposal path available for the spent fuel from the MSnB.

It will require stabilization prior to storage or disposal, which will require conversion to an inert borosilicate glass waste form unless an equally robust method of entombing the waste is developed. Once stabilized, the waste will be loaded into an appropriate MPC for dry storage pending the availability of a respository. Although the entire fuel salt mixture could be stabilized for disposal, it is desirable to recover the uranium and other valuable constituents, potentially significantly reducing the overall volume of waste. The trade offs here will be the economics of recovery, and the waste generated during these processes.

2.2.2 Transuranic Waste

Transuranic Waste (TRU) is a subset of the actinides, and consists of waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years. This type of waste is mostly unique to the DOE, as it is primarily a byproduct of nuclear weapons production. Because of the long half-lives of most of the TRU isotopes, this waste is also required to be disposed in a deep geologic repository.

The Land Withdrawal Act of 1982 established the Waste Isolation Pilot Plant (WIPP) as the national repository for defense related TRU, and established a similar oversight structure, with the EPA establishing the performance criteria, the DOE operating the facility, and the NRC certifying the confinement packages used during transport. The repository began accepting waste in 1999. There is currently no licensed repository for non-defense related TRU.

2.2.3 Low Level Waste

Low Level Waste (LLW) is essentially everything else. The Low-Level Radioactive Waste Policy Act of 1980 and the associated Amendments Act of 1985 gave States the responsibility to dispose of low-level radioactive waste generated within their borders and allows them to form compacts to locate facilities to serve a group of States, again regulated by the NRC for nuclear elements, and the EPA for other environmental issues. The primary laws which govern the environmental issues consist of the Resource Conservation and Recovery Act (RCRA) of 1976, which establishes the framework for the management of hazardous and non-hazardous solid waste. The Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) of 1980, known also as Superfund, is very similar to, and overlaps with RCRA, but is targeted at the cleanup of sites that were established before RCRA came into existence. The Toxic Substances Control Act (TSCA) governs the creation and handling of chemicals. The Clean Air and Clean Water Acts govern pollutant discharges to the respective mediums. All of these acts contribute to the requirements associated with establishing and operating a nuclear waste disposal site.

Because LLW is such a broad category, it is further subdivided into the following categories based on isotopic quantities, activities and half-lives:

a. Class A, which allows for low levels of radiation and heat, such that no shielding is required to protect workers or the public. Radioactivity in this category is expected to return to near background levels within 100 years of burial. This waste form is not required to be stabilized, and is expected to degrade significantly with time (e.g., organic materials such as paper, wood, or rags).

b. Class B, which allows for higher concentrations of radioactivity and requires greater isolation, packaging, and shielding than Class A, but is comprised of the same types of materials.

c. Class C, which is expected to decay to near background within 500 years, but requires isolation from the biosphere during this time due to mobility concerns. It must be buried at a depth exceeding 5 meters below the surface, and must have an engineered barrier separating the waste from the biosphere.

d. Greater than Class C (GTCC), which is not expected to decay to near background within 500 years. Typically, this waste is the commercial version of TRU.

The term near background is used, as all of these designations allow for some activity associated with isotopes of long half-lives. This is necessary as essentially all nuclear waste involves the combination of many isotopes which are impractical to separate, meaning short-lived isotopes are comingled with long-lived isotopes.

Under 10CFR61, each disposal site is required to evaluate the performance characteristics of the site, and develop a set of engineered barriers necessary to contain the waste for the specified time frame. These performance assurance measures are somewhat unique to each site as the weather, topography, geology, ground water, and potential intrusion scenarios are all somewhat unique. Similarly, a WAC is developed for each disposal site to ensure that every package placed within the repository complies with the repository performance design criteria.

There are currently 4 operating commercial LLW disposal sites in the US. One in Barnwell, SC, one in Andrews, TX, and one in Richland, WA, all of which are licensed for A, B, and C class waste. A facility located in Clive Utah is licensed for Class A waste only. There are also 2 government LLW disposal sites. One is located at the Nevada National Security Site (NNSS), and one at the Hanford site, both licensed for Class A, B, and C waste. No site is currently licensed for GTCC waste.

CHAPTER 3

A Description of the Molten Salt Nuclear Battery (MSnB)

The University of Idaho Nuclear Engineering Department has been developing the design of a small MSnB since 2017 as part of the graduate student program. The design evolves each year as refinements and further evaluations are performed.

The current design provides 400 kWt of continuous power for a period of up to 10 years, operating at or near atmospheric pressure. It uses a eutectic fluoride salt mixture composed of 46.5 molar percent Lithium Fluoride, 11.5 percent Sodium Fluoride, and 42 percent Potassium Fluoride (referred to as FliNaK). Uranium Fluoride is added to the salt at a ratio of 18 molar percent, and the uranium is enriched to 19.75 percent.

The reactor design is a compact series of nested right circular cylinders. The inner most cylinder is the reactor chamber, surround by a thick cylinder of beryllium oxide as a reflector. Within the reflector are 8 cylindrical control drum mechanisms which surround the core radially, evenly spaced from each other. Each control drum is composed of approximately 2/3 beryllium oxide, and 1/3 boron carbide. The geometry is such that when the drums are rotated in place, the sections of boron carbide are rotated closer to the core to reduce reactivity, and further away to increase reactivity. The reactor vessel is the outer most cylinder, surrounding the other components as well as providing the outer boundary of the fuel/salt flow channels.

The primary loop (molten salt) is physically configured as a torus, with the reactor chamber in the center, and the heat exchanger directly above the reactor chamber. The molten salt flows up from the reactor chamber, through the chimney, outward past the heat exchanger, down the outside of the reflector assembly, and back up into the reactor chamber. Heat is removed from the molten salt via an aluminum finned-tube heat exchanger. Heat is transferred to the heat exchanger via conduction and convection with the salt, through the heat exchanger wall by conduction, and from the fins to the gas secondary fluid loop (currently designed as air) via convection. This arrangement provides physical separation of the primary and secondary loops. The density difference of the salt resulting from the removal of heat is sufficient to circulate the salt, eliminating the need for pumping systems.

CHAPTER 4

The End-of-Life Radiological Source Term

The end-of-life source term is composed of three constituents. The first is the fission products, created by the act of fission. The second is the actinides, created by neutron absorption in the fuel which does not directly result in fission. The third is the activation products, created primarily by thermal neutron absorption in the non-fissionable materials that surround the reactor chamber.

4.1 Fission Products

The portion of the U235 which fissions is replaced by a host of intensely radioactive fission products. The primary issue associated with fission products is the amount of shielding necessary to protect personnel during reactor operation, reactor recovery operations, and to transport the spent fuel mixture for either reprocessing or long term storage. A secondary issue is how mobile the longer lived isotopes are in the environment, and how that impacts the disposal options.

The specific yields of the fission products are a function of the particular isotope being fissioned, as shown in figure 3.1. There are a number of sources which predict the yield of each isotope. For this analysis, I used the output of a Serpent code simulation of this specific reactor system developed during the NE 575 class by John Carter [12]. The computer simulation calculated the cumulative fission yields for 187 different fission products. Using these yields, I calculated the activity of each isotope at the end of the reactor life assuming a 400 kW(th) power level for 10 years of constant operation.

The amount of a particular fission product present at any moment in time is a function

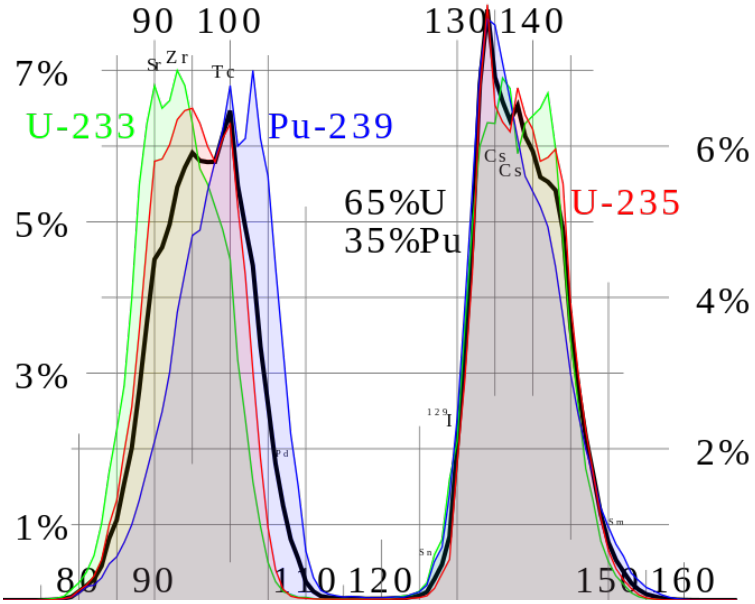


Figure 4.1: Fission Product Yields as a function of fissile isotope

of its production rate minus its decay rate:

$$d(N_x)/dt = F_s Y_x - \lambda_x N_x$$

Where N_x is the number of atoms of a given fission product, Y_x is the cumulative yield of that fission product, F_s is the number of fissions per second during reactor operation, λ_x is the decay constant for the fission product, and t is the time of reactor operation. This is a simplified model, as it assumes each fission product is generated at a constant rate according to its cumulative yield. The actual situation is more complex, as reactor operation is not at a constant power level, but this approximation provides a bounding simplification.

$F_s Y_x$ is a constant during reactor operation at full power, so the equation becomes:

$$d(N_x)/dt = C - \lambda_x N_x$$

The solution to this first order differential equation is

$$N_x(t) = C_1 e^{-\lambda_x t} + C/\lambda_x$$

$$N_x(t) = C_1 e^{-\lambda_x t} + (F_s Y_x)/\lambda_x$$

At t=0, $N_x = 0$, so C_1 can be determined

$$0 = C_1 + (F_s Y_x)/\lambda_x$$

$$C_1 = -(F_s Y_x)/\lambda_x$$

Substituting this value back in

$$N_x(t) = -(F_s Y_x)/(\lambda_x) e^{-\lambda t} + (F_s Y_x)/\lambda_x$$

Which simplifies to

$$N_x(t) = (F_s Y_x)/(\lambda_x)(1 - e^{-\lambda t})$$

Since the activity of a given isotope is the decay constant times the number of atoms, the activity for each fission product during the reactor operating period is given by

$$A_x = F_s Y_x (1 - e^{-\lambda t})$$

When the reactor is no longer operating, the production term goes to zero, and the activity becomes

$$A_x = A_{x0}(e^{-\lambda t})$$

where A_{x0} is the activity of the fission product present when the reactor operations end. The calculations to determine the 30 day post reactor shutdown activity of all the fission products tracked by the Serpent code are shown in Appendix C.1.

Because the fission products tend to be neutron rich, they primarily decay by beta emission, and most do so quickly, with only a few transitions before they achieve stability. Of the 187 fission products, 158 have half lives measured in days or weeks, such that by 30 days after reactor shutdown, they have effectively disappeared. Of the remaining 29, 13 disappear within a decade, 5 more disappear within a millennium, leaving only 4, as shown graphically in Figure 4.2. Of those 4, Rb-87 has such a long half live as to be effectively stable. Remarkably, only Tc-99, Zr-93, and Cs-135 require long term isolation from the biosphere.

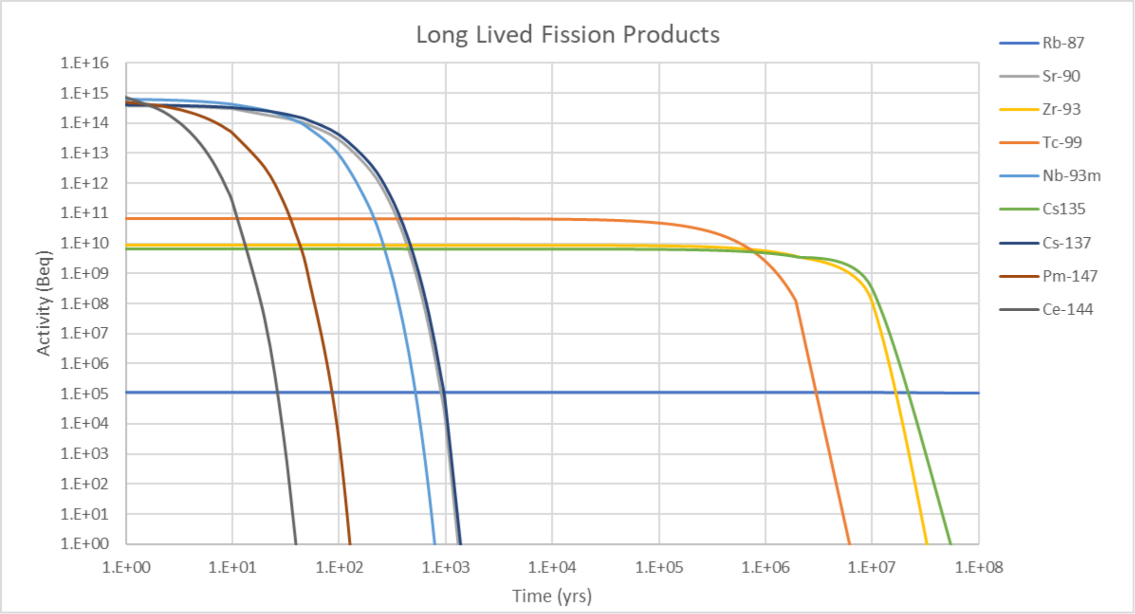


Figure 4.2: Fission product decay times

4.2 Actinides

Quantifying the actinides is a complex process, as they are created by successive neutron absorptions, and each one may decay by some combination of fission, beta, alpha, gamma, and neutron emission as shown in Figure 4.3. I used the Serpent model predictions of

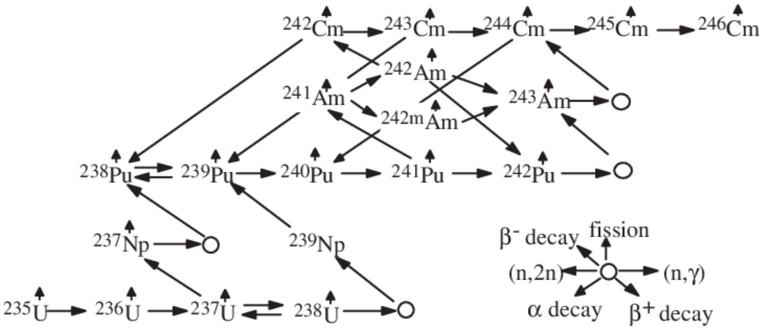


Figure 4.3: Actinide production and decay

each of the actinides at the end of reactor life, summarized in Table 4.1. As expected, Pu239 is the predominant actinide, as it results from a single neutron absorption by the large population of U238 atoms. Over half of the Pu239 produced is fissioned, but it still

Isotope	Activity
Pu 239	2.18E+11
Pu 240	1.65E+10
Pu 238	7.48E+09
Am 241	1.98E+09
Np 237	2.08E+07
Cm 244	2.66E+05
Cm 243	1.81E+05
Pu 242	1.26E+05
Am 243	5.33E+04
Np 235	1.08E+04
Np 236	1.13E+02
Pu 236	4.53E+01

Table 4.1: Actinides present in spent fuel

accounts for about 90 percent of the end-of-life actinide activity. The others are at least an order of magnitude less, as they require subsequent or multiple neutron absorptions. In general, as the path to creation becomes more complicated, the number of atoms generated becomes less, which is the reason the curium and californium isotopes are present in negligible quantities.

Actinides are the primary reason HLW requires long term isolation from the biosphere. They not only have long half lives, but quite unlike the fission products, the actinide decay chains are complex, generating many other isotopes of varying half lives and decay energies. They also decay by alpha emission, which is problematic if this material escapes confinement and is then ingested or inhaled.

However, their activity level is several orders of magnitude lower than the activity of the shorter lived fission products, and about the same as the activity of the longest lived fission products.

4.3 Activation Products

Activation occurs when the absorption of energy causes a non-radioactive atom to become radioactive. Although there are numerous types of reactions involved with acti-

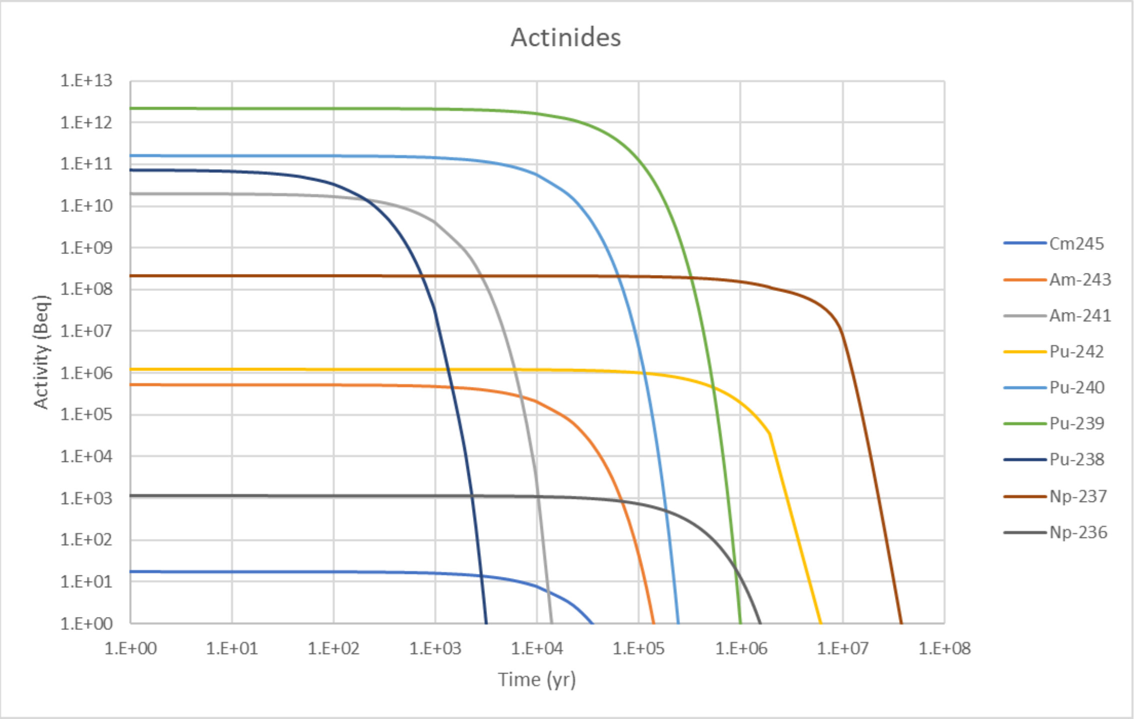


Figure 4.4: Actinide decay times

vation, this analysis will be limited to those isotopes which become activated by thermal neutron absorption, as that is by far the dominant mechanism by which most activation products are created. In addition, by assuming all fissions are caused by thermal neutrons, the number of activation products is maximized, making this a bounding analysis.

Activation is a function of the isotopic composition of the materials involved, the associated impurities within those materials, and the time and intensity of the neutron exposure. The number of atoms of a given activation product at the end of reactor life is a function of how many were created and how many have decayed. The rate of change in the number of such atoms is determined by

$$d(N_x)/dt = \sigma N_p \phi - \lambda_x N_x$$

and the activity of a given activation product at time t is given by

$$A_x = N_x \lambda_x = \sigma N_p \phi (1 - e^{-\lambda_x t})$$

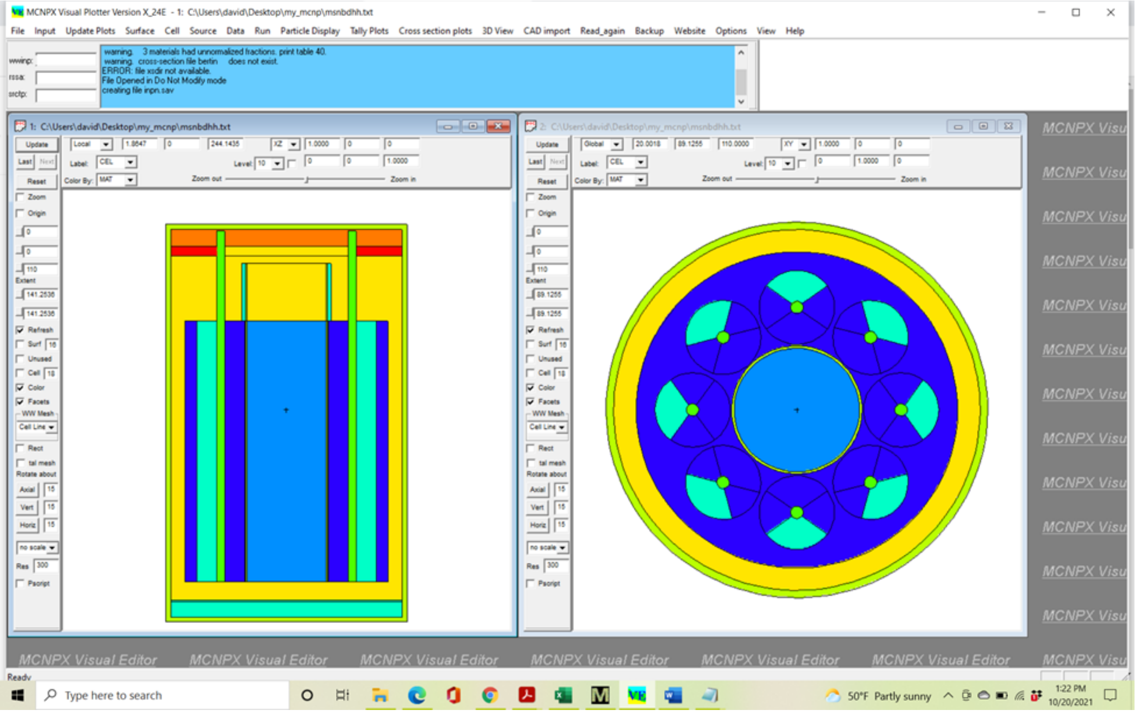
where N_x is the number of activated atoms of nuclide x, σ is the cross section for neutron absorption in the parent nuclide, N_p is the number of atoms of the parent nuclide, ϕ is the neutron flux, t is the time, and λ_x is the decay constant of nuclide x. The determination of A_x is problematic because the flux ϕ varies throughout the reactor vessel, and is difficult to determine as it is a function of neutron energies, vessel geometry, and the materials through which the neutrons pass.

4.3.1 Reactor Model

In order to obtain approximations of the flux at various points within the reactor vessel, I modeled the reactor in MCNP. I used the geometries provided in the NE 575 class drawings (FLINAK reactor drawings iteration 2-27-2021), as well as other information from the class design report. Figure 4.5 provides an illustration of the model. The MCNP card deck (file msnbdhh.txt) is provided in Appendix A to this document.

The reactor uses a proprietary graphite device within the reactor chamber to aid the natural circulation. Rather than trying to model that, I increased the enrichment within the core section of the reactor to account for the increased moderation of the graphite until the system achieved criticality. This occurred at just over 32% enrichment within the core.

I used the f4 tallies to quantify the average neutron flux within a cell. MCNP tracks particles through cycles, and therefore does not report the neutron flux in units of $n/(cm^2 s)$, but rather in terms of n/cm^2 per source neutron. To account for time, the tally results must be multiplied by the number of source neutrons per unit time. At a reactor power level of 400 kW(th), the number of source neutrons per unit time is obtained



```
warning.    3 materials had unnormalized fractions. print table 40.
warning.  cross-section file bertin     does not exist.
ERROR: file xsdir not available.
File Opened in Do Not Modify mode
creating file inpn.sav
```

Figure 4.5: MCNP model of the MSnB

from

$$(400000J/s)(1MeV/1.602(10^{-13})J)(f/200MeV)(2.43n/f) = 3.03(10)^{16}n/s$$

The tally results are provided in Table 4.2. Using the flux values produced by the model, the individual activation products can be quantified. There are numerous calcu-

Component	MCNP Cell	Volume	n/cm^2	n/cm^2s
Core	2	3.28E5	1.39E-4	4.21E12
Hastelloy rod	7	4.42E3	2.56E-5	7.76E11
Outer SS vessel	43	4.59E5	4.27E-6	1.29E11
Absorber base plate	45	1.67E5	1.72E-5	5.21E11
internal 304 plate	46	1.67E4	2.53E-5	7.67E11
Chimney wall	47	5.93E3	4.56E-5	1.38E12
Reactor chamber wall	49	2.68E4	1.31E-4	3.97E12
Upper absorber	50	1.26E4	4.38E-5	1.33E12
External Concrete Shield	54	6.40E7	2.14E-7	6.48E9

Table 4.2: Neutron Flux Values Experienced by Various Reactor Components

lations involved, as the reactor has components of 304 stainless, Hastelloy, boron carbide, beryllium oxide, and graphite, and the reactor external shield is concrete. The process used to quantify the activation is the same for all materials, and consists of the following steps:

1. The atomic constituents, expected impurities, and relative percentages were obtained by a review of the material specifications provided by material fabricators or ASTM material standards;
2. The time of neutron exposure was assumed to be 10 years;
3. The amount of each material was taken from the reactor system drawings or the MCNP model;
4. The cross sections for neutron capture were taken from the International Atomic Energy Agency (IAEA) Atlas of Neutron Capture Cross Sections;
5. The magnitude of the neutron flux was obtained from the MCNP model;
6. The abundance of each isotope within each material was obtained from the IAEA Live Chart of the Nuclides, as was the decay chain and associated daughter products.

The process as applied to graphite is presented in the following text, as it is the simplest of the reactor materials. All other activation calculations are summarized in the text, and detailed in Appendix B to this document.

4.3.2 Graphite

Figure 4.5 illustrates the carbon activation chain. Typically at each step the number of atoms decreases, such that from beginning to end, the difference is many orders of magnitude, and eventually the chain ends. The black indicates stable isotopes. The orange indicates isotopes which have short enough half lives that they will disappear once the reactor shuts down. The yellow indicates the isotopes which will affect transportation and disposal. Blue arrows indicate the path following thermal neutron absorption, while yellow and orange arrows indicate the path following decay, as well as identifying the

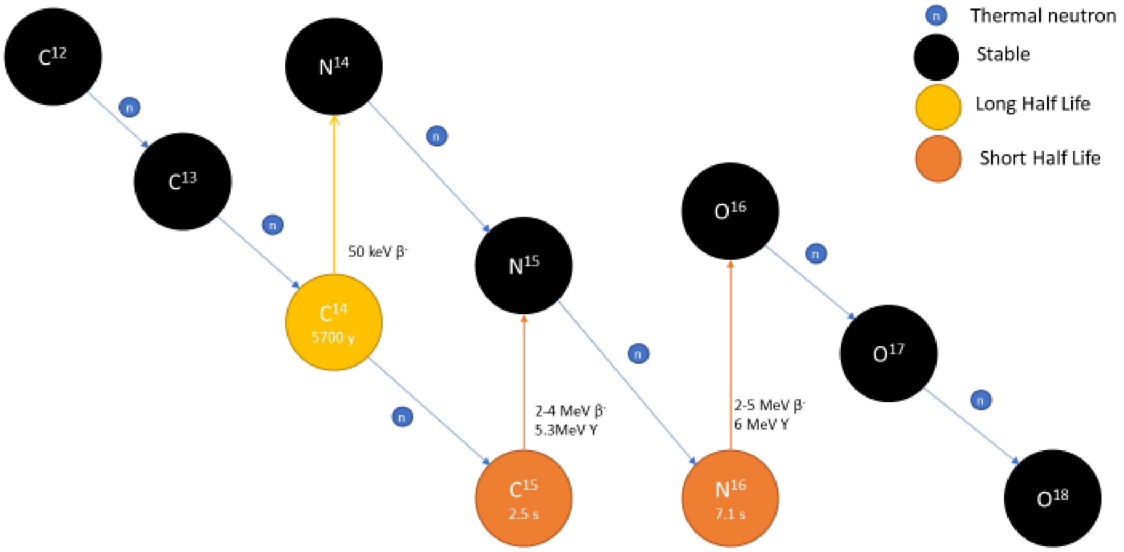


Figure 4.6: Sequence of graphite activation

energies and particles associated with decay.

As it occurs in nature, graphite is composed of two isotopes, C12 at 98.9% and C13 at 1.1%, and has a Molecular Meight (MW) of 12.01 g/mole. Various manufacturing processes can routinely deliver carbon which is 99.9% pure. The typical impurities are the transition metals, so a wide variety of impurities exist. Even at the low value of 0.1%, these can have significant effects. The activation impacts from impurities for all materials will be discussed separately.

The MSnB design uses 315 kg of graphite inside the reactor chamber as a device to aid the natural circulation, so the chain begins with 1.58E28 atoms of carbon (mostly C12). Per the model, the thermal neutron flux in this area is $4.21(10^{12})n/cm^2s$. Per the IAEA's atlas of thermal neutron cross sections, the absorption cross section for C12 is about $7(10)^{-2}b$. Therefore, the amount of C12 which will convert to C13 via neutron absorption over the 10-year operating period of the reactor is

$$N_{C13} = N_{C12}\sigma_{C12}\phi t = (N_A m_{C12}/MW_{C12})\sigma_{C12}\phi t$$

$$N_{C13} = (6.02(10)^{23}(3.15(10)^5)(0.989)/12.01)(7(10)^{-26})4.21(10^{12})(3.15(10)^8)$$

$$N_{C13} = 1.45(10)^{24} atoms$$

C13 is a stable isotope, so there is no decay component. Conservatively, this quantity is then added to the natural abundance of C13, and the amount of C13 which converts to C14 is similarly determined as

$$N_{C14} = N_{C13}\sigma_{C13}\phi t = (N_A m_{C13}/MW_{C13} + 1.45(10)^{24})\sigma_{C13}\phi t$$

$$N_{C14} = (6.02(10)^{23}(3.15(10)^5)(0.011)/12.01)((10)^{-27})4.21(10^{12})(3.15(10)^8)$$

$$N_{C14} = 2.33(10)^{20} atoms$$

C14 is radioactive, with a half live of 5700 years. It decays entirely by a 50 keV β^- emission into N14, which is stable. Some of the C14 will decay into N14:

$$N_{N14} = N_{C14}(-\lambda_{C14})(t) = 2.33(10)^{20}3.86(10)^{-12}3.15(10)^8 = 2.83(10)^{17}$$

some of which will activate into N15:

$$N_{N15} = 2.83(10)^{17}(8(10)^{-26})4.21(10^{12})(3.15(10)^8) = 3.00(10)^{13} atoms$$

Because it has a long half life, some of the C14 will activate to C15:

$$N_{C15} = 2.33(10)^{20}(1(10)^{-30})4.21(10^{12})(3.15(10)^8) = 3.09(10)^{11} atoms$$

C15 is also radioactive, but has a very short half live of 2.5 seconds, so essentially immediately decays by emission of a high energy (2-4 MeV) β^- and a high energy (5 MeV) gamma into stable N15. Some of the N15 will convert to N16:

$$N_{N16} = (3.00(10)^{13} + 3.09(10)^{11})(8(10)^{-28})4.21(10^{12})(3.15(10)^{8}) = 3.22(10)^{7} atoms$$

N16 has a 7 second half life, and decays via emission of high energy β^- and γ into stable O16. Some of the O16 converts to stable O17:

$$N_{O17} = 3.22(10)^{7}(9(10)^{-27})4.21(10^{12})(3.15(10)^{8}) = 385 atoms$$

terminating the chain.

From an activation standpoint, graphite is an excellent material, as even under the intense neutron flux of the core, it produces only C14 as an activation product which remains at the end of reactor life. For this specific reactor, the end of life activity of the C14 is

$$A_{C14} = N_{C14}\lambda_{C14} = 2.33(10)^{20}3.86(10)^{-12} = 8.97E8Beq$$

or about 0.024 Curies. In larger quantities, C14 can be a disposal issue because of its long half live and high mobility in the environment, but the low energy beta poses little dose risk.

4.3.3 Beryllium Oxide

Beryllium oxide is the reflector material surrounding the core. Beryllium is 100% Be9, while oxygen is over 99% O16, with traces of O17 and O18. The activation chains for the two components are shown in Figures 4.7 and 4.8. The oxygen chain produces no long term activation products. The beryllium chain only produces Be10, a long lived Beta emitter which decays to stable B10. Because of the very long half life of Be10, only a small portion of the Be atoms become B10. However, B10 has a very large cross section for thermal neutrons (3813 barns), meaning that essentially all of the B10 is converted into Li7 and an alpha particle (the green arrows in Figure 4.8). Some of the Li7 will activate into Li8, which immediately decays into 2 high energy alpha particles and a large gamma. The alpha particles will pick up electrons and become helium atoms. Thus, although activation per se will not be a large problem in the beryllium oxide, the helium will accelerate both swelling and corrosion problems. The reactor contains $4.54(10)^6$ g of beryllium oxide, which translates to $2.19(10)^{29}$ molecules at the beginning of reactor life. From this, less than 0.1 curies of Be10 will be created, and $9.15(10)^{17}$ alpha particles.

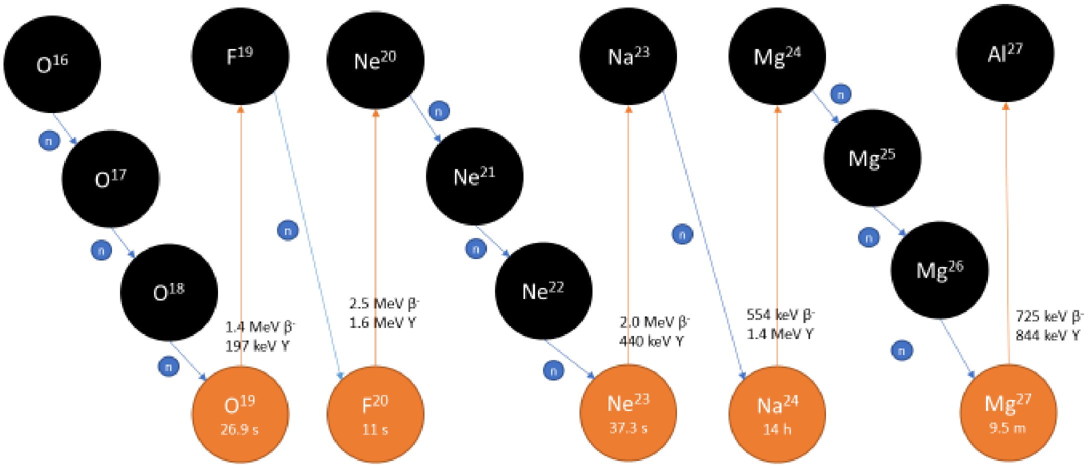


Figure 4.7: Oxygen Activation Chain

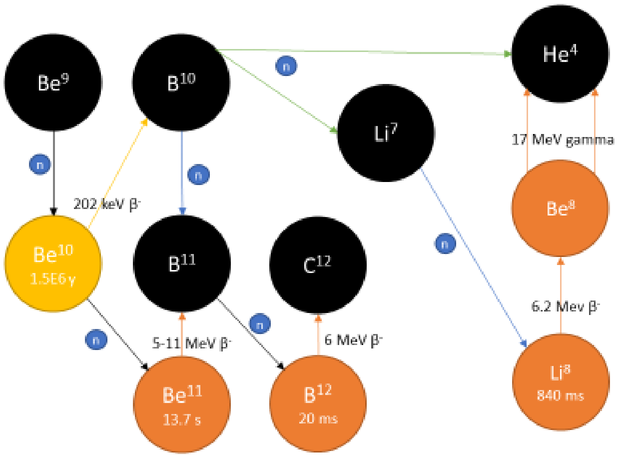


Figure 4.8: Beryllium and Boron Activation Chain

4.3.4 Boron Carbide

The boron carbide is used as the neutron poison in the control drums. The reactor contains $1.44(10)^6$ grams of boron carbide. The only activation product created in the boron carbide is a small amount of C14. However, because boron contains 20% B10, the production of helium is much higher than in the beryllium oxide, and results in $1.78(10)^{23}$ alpha particles, or about 1/3 mole of helium.

4.3.5 Aluminum

Aluminum is used in the heat exchanger to transfer heat from the molten salt to the secondary fluid (air). The aluminum chain produces no long term activation products, as shown in Fig 4.9.

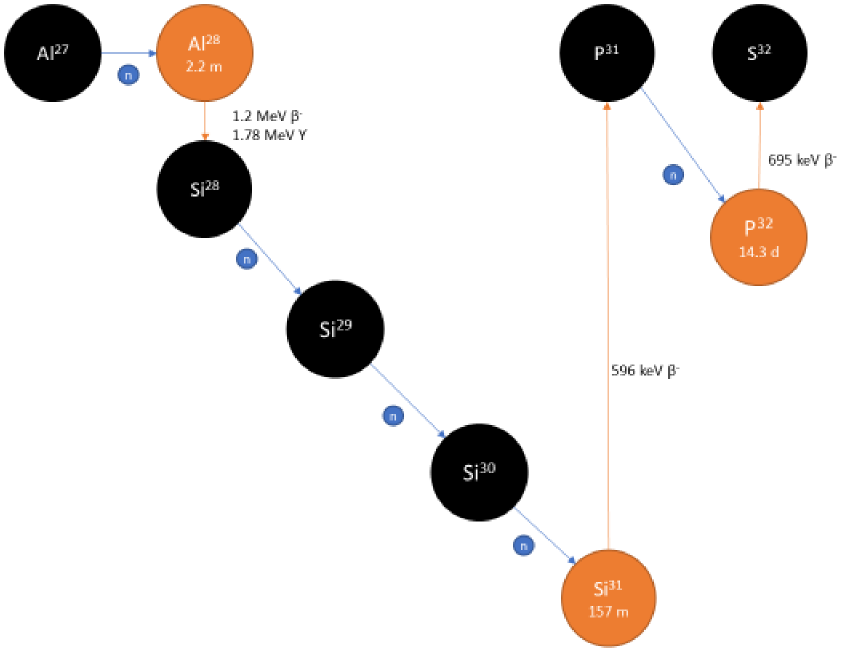


Figure 4.9: Aluminum and Silicon Activation Chain

4.3.6 Hastelloy

Hastelloy is used as the pivot rods for the control drums. The reactor contains $1.95(10)^5$ grams of hastelloy within the 8 pivot rods. Hastelloy consists of about 5% iron, 16.5% molybdenum, 7% chromium, and the balance of nickel, with a host of minor contaminants. The activation chains are shown below.

The iron activation chain produces Fe55. While Co60 appears at the end of the chain, it is so far down the chain that essentially no Co60 is produced.

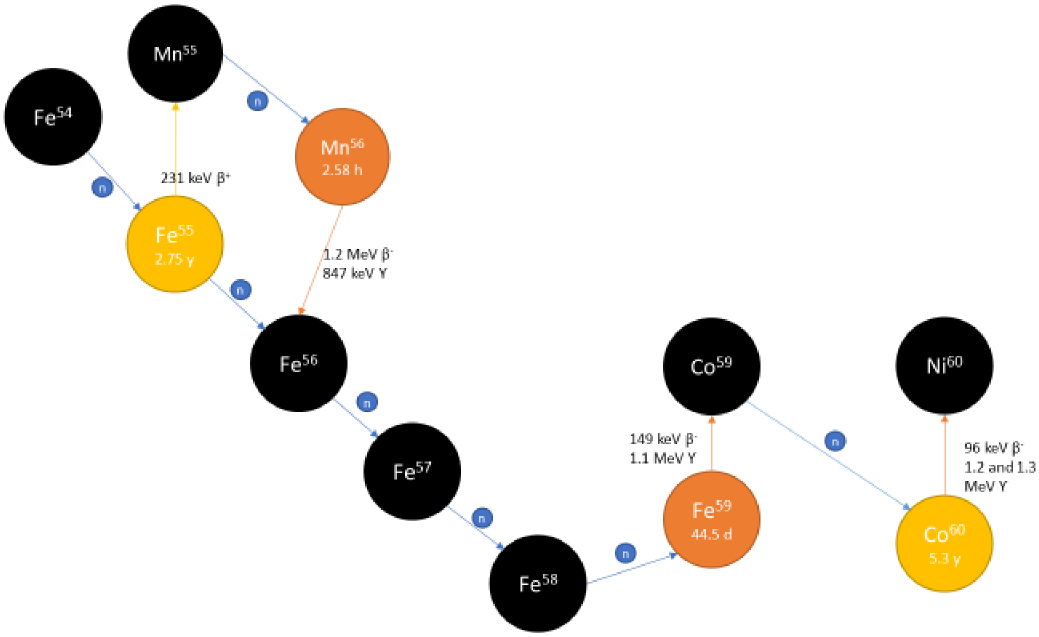


Figure 4.10: Iron Activation Chain

The molybdenum activation chain is more complex, as there are 6 naturally occurring isotopes. It produces Mo93, Nb94, and Tc99. Very little Nb94 is produced, as Mo92 is 14.53% abundant, Mo93 has a long half life, and both have relatively low thermal neutron capture cross sections.

Tc99 is a potential problem, as it is a relatively high yield fission product, forms insoluble fluorides with the molten salt, is an activation product, and is highly mobile in the environment.

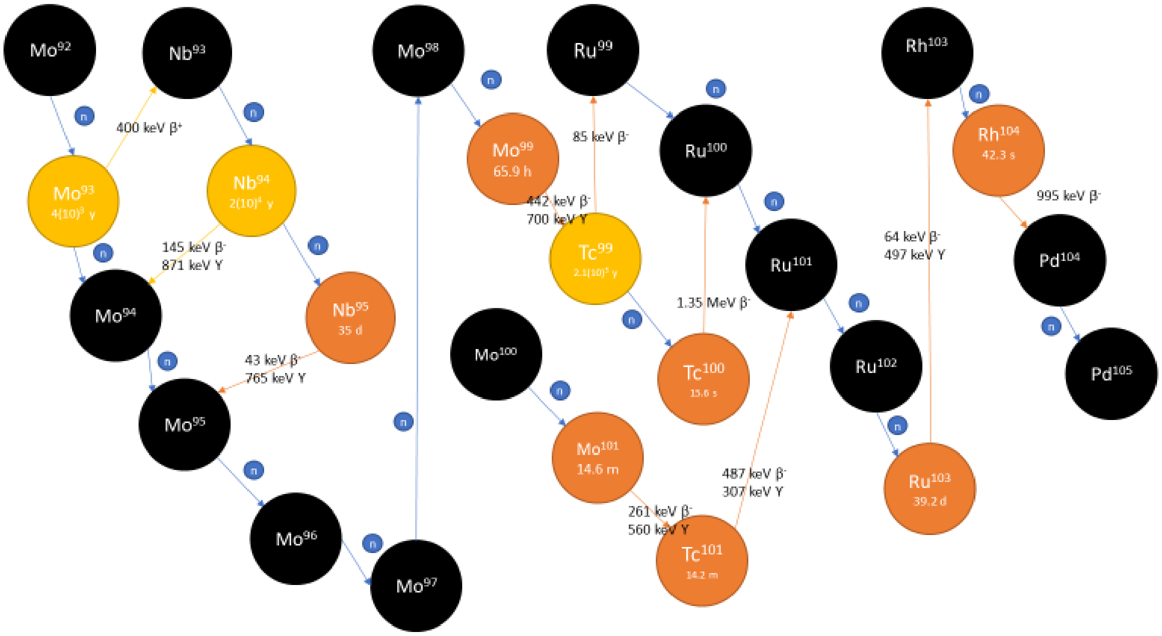


Figure 4.11: Molybdenum Activation Chain

The chromium chain begins with Cr50, which is not technically a stable isotope, but has such a long half life that it still accounts for 4.3% of naturally occurring chromium. This chain includes the Manganese chain as well, and reenters the iron chain. Chromium produces no long lived activation products.

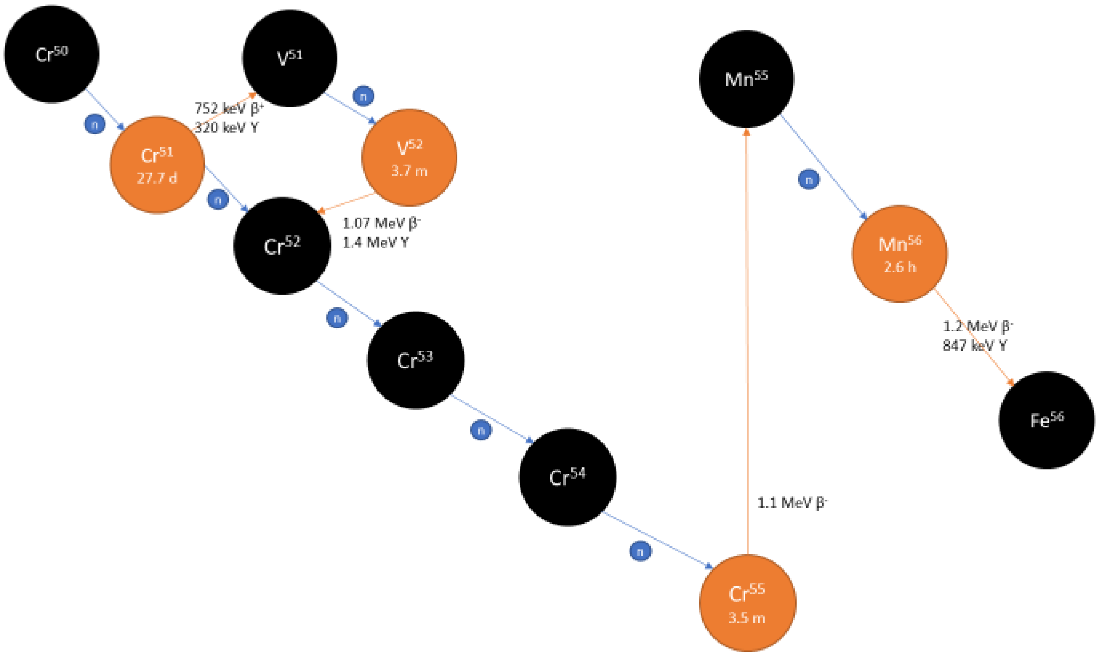


Figure 4.12: Chromium Activation Chain

The nickel chain produces Ni59 and Ni63. Very little Co60 is produced in this chain, for the same reasons as the Nb94 in the molybdenum chain.

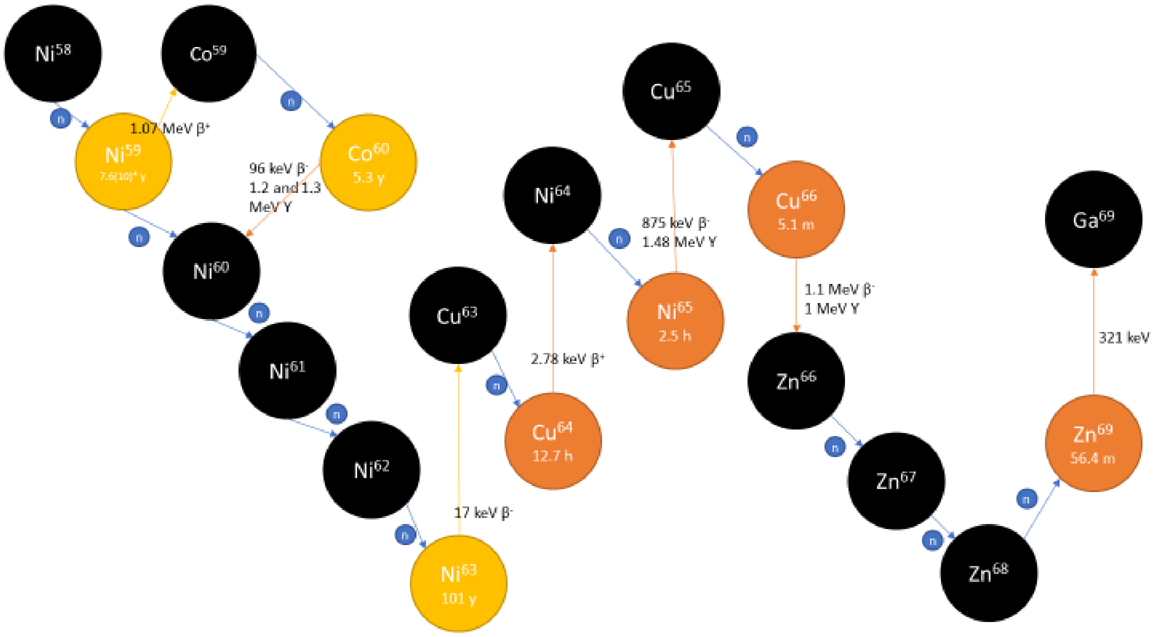


Figure 4.13: Nickel Activation Chain

4.3.7 304 Stainless Steel

304 SS is the primary structural material within the reactor, used to form the inner and outer reactor vessels, as well as various plates and containers. The reactor contains $4.66(10)^6$ grams of 304. It consists of about 20 percent Chromium, 11 percent Nickel, and the balance of Iron. Therefore, the only activation products produced by the base materials within the 304 are Fe55, Ni59 and Ni63.

4.3.8 Impurities

Impurities are of course present in every material, but particularly in the steels. Common contaminants include Phosphorus, Sulfur, Silicon, Tungsten, Titanium, Copper, Manganese, Carbon, Zinc, and Cobalt. The activation chains for these elements which are not presented in the text are provided in Appendix B1. None of them produce long term activation products with the sole exception of cobalt, which produces Co60. Even in minor concentrations, Co60 is a serious problem for 4 reasons. First, all natural cobalt is Co59, so a single neutron absorption creates Co60. Second, Co59 has a large cross section of 37 barns for the absorption of thermal neutrons. Third, every decay of a Co-60 atom emits 2 consecutive gammas of 1,173 and 1,332 keV. Fourth, the half life of Co60 is only 5.3 years, so the activity is high. Because of this, Co-60 is a major concern for dose rate in the spent reactor vessel, and therefore primarily a reactor maintenance and transportation issue. I used the lowest values of Co59 contamination I could find, which was 0.05% in the 304 and 0.1% in the hastelloy. The hastelloy is higher because it has more nickel, and cobalt is found in higher concentrations in nickel based ores.

In the first generation of reactors discussed at the beginning of this thesis, there were no viable alternatives to stainless steels, and in fact Hastelloy was specifically developed for the Molten Salt Experimental Reactor constructed at Oak Ridge in the 1960s. The intense radiation fields associated with Co60 have been the driving reason that maintenance, decommissioning, and deactivation activities for all PWRs and BWRs are so expensive and time consuming. The short half life of Co60 is why it was considered a good idea to let a plant sit idle for several years before attempting decommissioning.

4.3.9 Concrete

Concrete is used as the external shield to the reactor vessel, as well as a platform for the reactor components. All concrete requires internal reinforcement to provide strength and stability. Reinforcement steel Bar (rebar) is normally used, but the steel adds significantly to the activation of the concrete. There are alternatives to rebar, such as glass fiber, carbon fiber, basalt fiber, and woven strand bamboo. All of these also affect activation, but none contain cobalt, making the impact far less. For simplification, I have ignored the reinforcement in this analysis, and just addressed ordinary concrete.

Ordinary concrete is a mixture of H, C, O, Mg, Ca, Na, Fe, Al, Si, and K. The only new elements beyond those previously discussed are H, Ca, and K, whose activation chains are shown in Figures 4.14 and 4.15. Per the figures, concrete activation products include H3, C14, Ca41, Fe55, and K40.

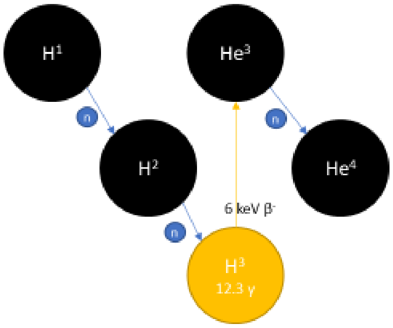


Figure 4.14: Hydrogen Activation Chain

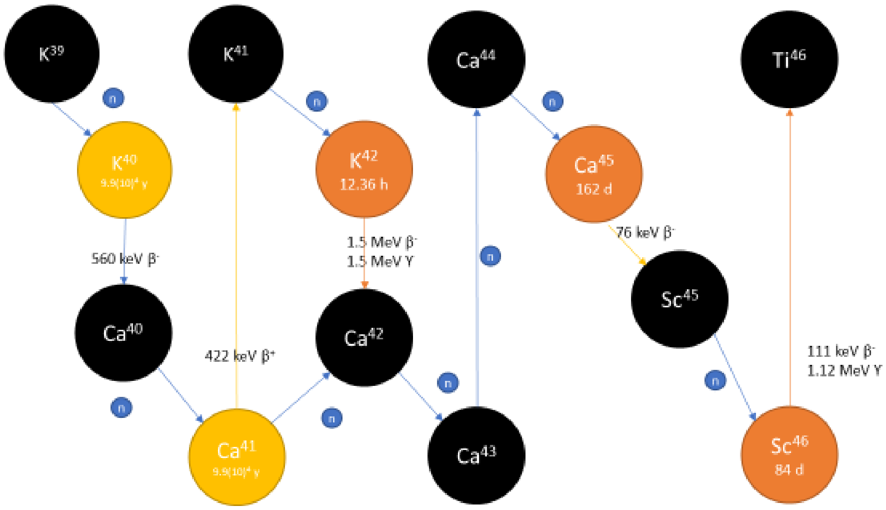


Figure 4.15: Calcium and Potassium Activation Chain

4.3.10 Reactor component summaries

From the above discussion, there are twelve long term activation products within the MSnB. The calculations to determine activity are provided in Appendix B2, and summarized in Table 4.3.

	Inner Vessel 304SS	Chim-ney 304SS	Outer Vessel 304SS	Bottom Plate 304SS	Pivot Rods Hastelloy	Reflect Be Oxide	Abs B_4C	Shield Conc
Co60	6.08E3	5.24E2	1.89E3	4.11E2	1.96E3			
Fe55	5.11E4	3.94E3	1.42E4	3.09E3	3.15E2			9.93E1
Ni59	1.15E1	8.89E-1	3.21	6.98E-1	1.5E1			
Ni63	1.40E3	1.08E2	3.9E2	8.47E1	1.82E3			
Tc99					7.63E-2			
Mo93					1.1E-2			
Nb94					3.33E-11			
Be10						5.21E-1		
C14							8.55E-1	2.15E-5
H3								8.69E-3
Ca41								2.9E-1
K40								2.13E-5

Table 4.3: Activation product summary (Ci)

4.4 Dose Rate

4.4.1 Fission products

Because fission products are neutron rich, they usually decay by a series of beta emissions, most of which also emit gammas and X-rays. Some emit neutrons as well, but the half lives of these are nearly all less than a minute. As a consequence, the dose rate after reactor operation has ceased is primarily from beta/gamma emission, so the required shielding is a function of the gamma emission strength.

To estimate the dose rate associated with the spent mixture as a function of time, I took the end of life activity of each fission product, and applied its specific exposure rate constant to determine the dose at a given distance (assumed to be 1 meter):

$$DR = A_x \Gamma_x / d^2$$

$$DR = A_{x0}(e^{-\lambda t})\Gamma_x / d^2$$

where Γ_x is the dose rate constant for a given fission product, and d is the distance from the reactor core. The dose rate calculations for the fission products present at 30 days post reactor shutdown is provided in Appendix C.2.

Of the 187 tracked fission products, 158 have half-lives of a few days or less, such that after 30 days, they are effectively no longer present. Of the remaining 29, only 11 contribute to the dose in magnitudes of 1 R/hr or greater, and only 4 are strong gamma emitters, with the rest primarily emitting betas, as summarized in the following table.

Isotope	R/hr	Gamma keV	Occurrence
Zr-95	17,049.39	750	98
Nb-95	13,406.28	766	100
Ru-103	4,350.11	500	91
Cs-137	3,542.58	662	85
Ba-140	1,060.45	537	24
I-131	403.93	364	81.5
Y-91	67.15	1,205	0.26
Xe-133	14.94	81	36.9
Y-90	7.14	0	0
Te-132	6.54	228	88
Mo-99	2.24	740	12.2

Table 4.4: Dose contributors in the fission products

Zr-95, Nb-95, and Ru-103 all have half lives in the 1-2 month range, where as Cs-137 has a 30 year half live. As a consequence, the gamma dose rate will decrease by an order of magnitude during the first year following shutdown, until the Cesium becomes the predominant gamma emitter. Figure 3.2 illustrates this decay, and highlights a major issue for small mobile reactors. At 30 days after shutdown, which is our target timeframe to transport the spent fuel, the gamma dose from the spent fuel salt mixture at 1 meter will be approximately 40,000 R/hr.

4.4.2 Activation Products

Because the activation products are locked inside the various materials, only the gamma components will contribute to the dose rate. For this reason, H3, C14, Be10, K40, Ca41, Fe55, Ni59, Ni63, Mo93, and Tc99 do not contribute. This is remarkable, as the only

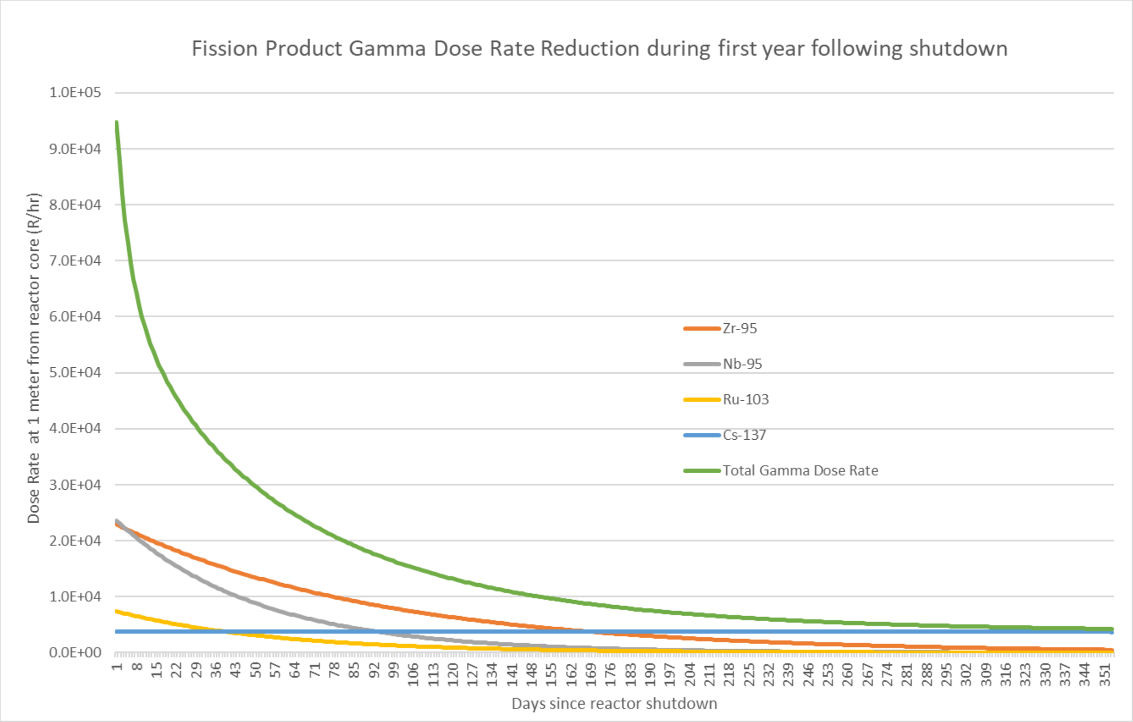


Figure 4.16: Fission Product Gamma Dose Rate

significant activation product which contributes to the gamma dose rate is Co60. The Co60 produced in the four 304 stainless steel components and the hastelloy in the control drum pivot pins amounts to $1.16(10)^4$ curies at the end of reactor life. The curie value is this high because Co60 is a very strong gamma emitter, in that nearly every decay of a Co-60 atom emits 2 high energy photons of 1,173 and 1,332 keV. Assuming a point source at the center of the reactor, this single isotope will cause a dose of

$$DR = A_x \Gamma_x / d^2 = 1.16(10)^4(1.29) = 1.4964(10)^4 R/hr$$

at 1 meter. I find it stunning that a minor contaminant in these materials, limited to less than a tenth of a percent can result in a gamma dose rate that approaches 40% of the dose rate produced by all the fission products remaining at 30 days post reactor shutdown.

4.5 Decay Heat

Decay heat has important ramifications for transport, as all Type B shipping packages are sealed (unvented) during transport. Heat build up within the package can distort sealing surfaces, resulting in breaching the package. To prevent this, wattage limits are applied to the payloads. The fuel/salt mixture is the only reactor component expected to ship inside a Type B package, so only the 29 tracked fission products remaining after 30 days will be considered in this evaluation. I calculated the decay heat using two different methods. The first was using the formula:

$$DH = A_x Q_x$$

where Q_x is the energy released with each decay. I obtained values for Q_x from the IAEA's Live Chart of the Nuclides. This calculation showed a value of 1,548 W at 30 days after shutdown. This is a bounding value, as the Q_x values assume the highest energy beta emissions with the associated gammas. The betas will convert entirely to heat since none escape the spent mixture, but some of the gammas will escape.

In the second method I used the empirically derived Wigner-Way formula:

$$DH = 0.0622 P_0 (\tau^{-0.2} - (\tau_0 + \tau)^{-0.2}))$$

where P_0 is the full thermal power output of the reactor, τ is the number of seconds since reactor shutdown and τ_0 is the number of seconds the reactor operated at power. Using 400,000W as P_0, 2.592E6 (30 days) as τ and 3.1536E8 (10 years) as τ_0, this provides a decay heat value of 801 Watts.

CHAPTER 5

Design Approach to Achieve Mobility

The design approach is based on making each component small enough to transport over the road, in a manner which requires no exemptions from current shipping regulations, with a minimum of specialized or heavy infrastructure (railways, facilities, high capacity cranes, etc.), and using currently available transport packages.

Disassembly of system components at the deployment location will be designed to be performed without exposing personnel to high radiological dose areas, and without the need for radiological contamination controls. Examples include quick disconnect mechanisms for piping, remotely operated valves, and engineered attachment devices to allow remote rigging of system components to lifting and handling equipment.

The deployment site will be prepared by excavating a space large enough to house the reactor vessel, the fuel/salt drain/transport tanks, and the external shielding.

The external shielding is installed first. It is designed to prevent activation of the surrounding earth, while allowing the earth to provide the gamma shielding during operation. The fuel/salt transport containers are installed next. The reactor vessel is then positioned above and connected to the containers to allow remote draining of the fuel/salt mixture from the reactor vessel at the end of use. The piping connections are then made between the reactor vessel and the heat extraction/power module. Finally, the electrical connections for operation and monitoring of the system are connected to the control module.

At end of use at a particular deployment location, the fuel/salt is drained to the tanks, and replaced in the reactor vessel with a high density grout, such as barite concrete. The grout is designed to fix radiological contamination in place within the reactor vessel, provide shielding of the internal activated components, and eliminate void spaces within the vessel. The reactor vessel is remotely separated from the tanks, and placed inside an IP-2 or IP-3 transport container, and shipped as a unit for disposal as LLW at a suitable location. For this analysis, the disposal location will be the DOE LLW site located in Nevada.

The spent fuel/salt tanks are then individually removed from the external shielding vault, and transported via a shielded Type B shipping package to a separate site for reprocessing or repackaging into an MPC for long term dry storage. For this analysis, the NAC-LTW Type B cask will be used for transport.

The external shield will be disassembled and shipped as LSA-III for either disposal as LLW or storage pending reuse.

Finally, the non-nuclear components are removed, the excavation filled in, and the site restored to its condition prior to deployment.

5.1 Modular Construction

Although the reactor vessel, even fully loaded with fuel would easily fit on a single semi tractor trailer assembly, the shipment of such an arrangement is prohibited by the transport regulations. In the case of fresh fuel, the problem is the potential for nuclear criticality during a transport accident. The use of molten salt as a carrier for the fuel makes this problem more difficult, as it will liquify during the fire accident scenario, significantly complicating the task of keeping the fuel subcritical. The spent fuel has the same problem, but is overwhelmed by the very large radiological dose rates associated with the end of life source term.

It is still possible to ship the entire vessel loaded with fuel, but this requires the use of a large Type B shipping package.

As noted earlier, the dose rate from the spent fuel is expected to be 40,000 R/hr, as well as 15,000 R/hr from the activated components, with a decay heat load between 800 and 1500 watts, and a fissile material loading of approximately 615 kg. Combined with the weight and size constraints, these amount to essentially worst-case transport conditions, and thus severely limit the available options for Type B packaging. Only the largest, most heavily shielded Type B packages will work, and these are all rail mounted, as the casks themselves weigh in the neighborhood of 200 tons.

The US Navy uses two such packages that have the capability to transport the reactor. One is the M140 cask, and the other is the M290 cask (shown in Figures 5.1 and 5.2).

Figure 5.1: Navy M140 spent fuel transport cask

Both are used to transport spent fuel from the shipyards to the examination facility in Idaho. The M290 is a dual-purpose cask, as it is also intended to carry spent fuel disposal containers to the yet to be established national HLW repository. It is capable of carrying a 96,000-pound payload in the form of a right circular cylinder, 66 inches in diameter and 185 inches in height. The fully loaded package weight is 260 tons. As the Navy owns all such railcars, their approval will be required to use the package and modify the package Certificate of Compliance to address our payload. The Navy is unlikely to provide such approval, but has provided DOE with design information for the development of similar DOE casks. The DOE is currently working on two such casks. One is an 8-axis railcar

Figure 5.2: Navy M290 spent fuel transport cask

(Fortis) and the other a 12-axis railcar (Atlas), both of which are similar in appearance to the M290 (see Figures 5.3 and 5.4). Because the MSnB will not fit closely inside the payload compartment, shock-absorbing internal spacers will need to be designed, tested, and certified through the NRC to prevent dynamic loading during potential accident scenarios. Importantly, the DOE packages are not expected to be certified and available until 5 to 10 years from now.

Use of one of these packages will require a main-line railway system to transport the reactor to and from the remote deployment location. A significant amount of high capacity lifting and handling equipment is required so that the Type B package can be disassembled, loaded, and reassembled. The external impact limiters must be removed and stored in a manner to prevent damage, the cask rotated to the vertical position, and possibly removed from the railcar and positioned in a stand. The cask lid must be removed and stored. The internal impact limiters are then removed so that the reactor can be loaded. The lower impact limiter is installed, followed by the reactor and then the upper internal impact limiter. The lid is then replaced, and a leak test performed to verify the integrity of the seals. The cask is then transferred back to the railcar, rotated back to the horizontal position, and secured in place. The external impact limiters are then reattached. Due to the high radiation dose rates, the lifting and handling of the reactor will have to be performed remotely, which will in turn either require a large purpose-designed facility, or be performed outside, where the operations will be subject to weather limitations such as wind and temperature.

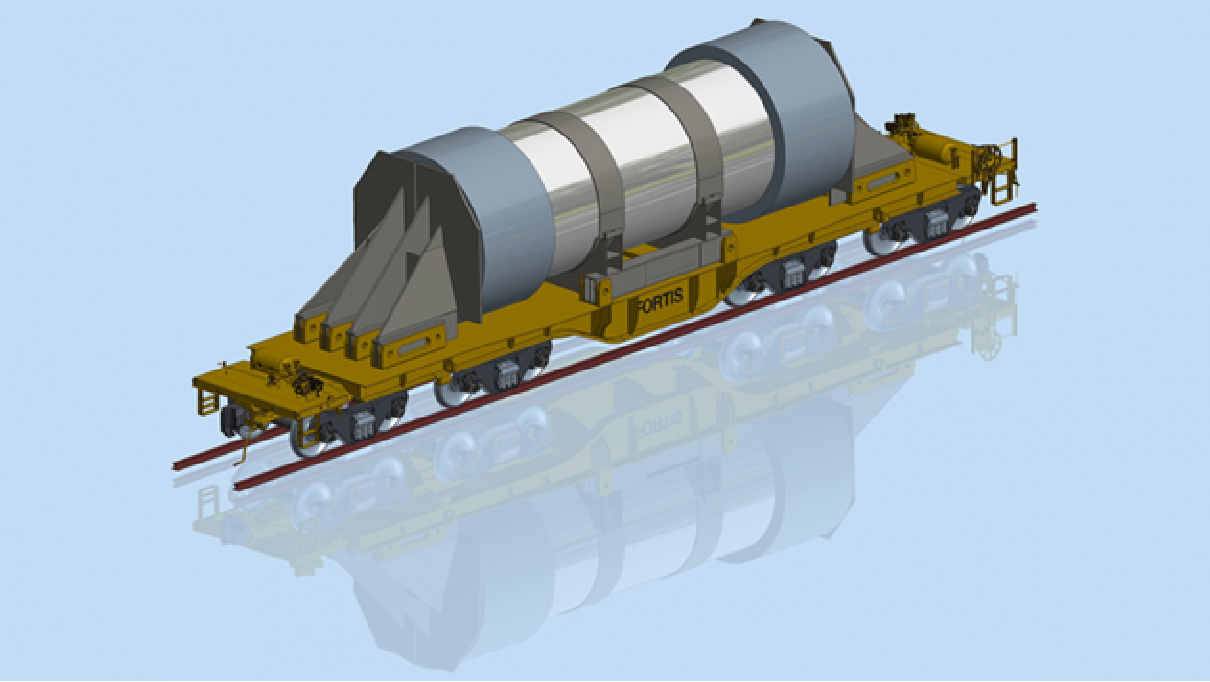


Figure 5.3: Future Fortis DOE spent fuel transport cask

Of primary importance is the fact that the reactor cannot be disposed as is, because the FLiNaK is not a stabilized waste form that meets the HLW repository WAC. If shipped as a single unit, regardless of where it is shipped, the reactor will have to be unloaded and disassembled into the various components so that the fuel can be reprocessed and the resulting waste stabilized, and the non-fuel components can be characterized and disposed. Therefore, transport of the reactor as a single unit offers no benefit, and introduces both significant cost and risk to timely reactor deployment and recovery. I therefore rejected this option in favor of shipping the individual reactor components. By separating the fuel from the reactor vessel, we can use currently available, standard commercial over-the-road

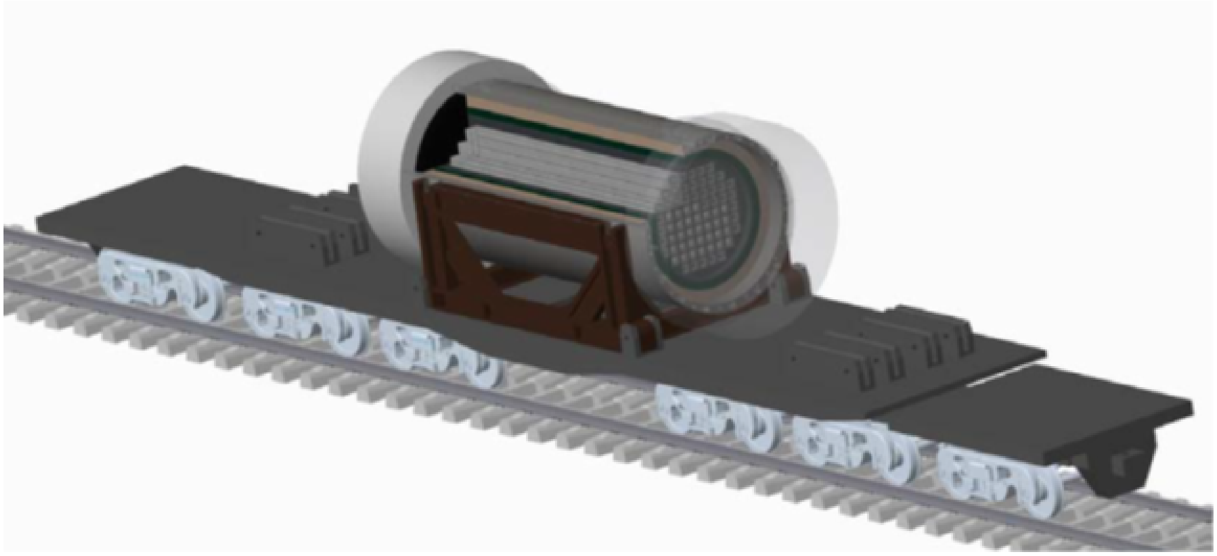


Figure 5.4: Future Atlas DOE spent fuel transport cask

shipping equipment, eliminating the need for nearly all infrastructure, and requiring only a road (or ice, or water) to access the deployment location.

5.2 Fuel/Salt Transport

The fuel/salt mixture at the beginning of reactor life consists of 2.596E6 grams (0.86 curies) of U-238, and 6.37E5 grams (1.34 Curies) of U235. Unirradiated enriched uranium can be shipped in either a fissile-qualified Type A package (Type AF) or a Type B package.

The concern with this source term is to ensure the fissile material remains both subcritical and contained during transport to the reactor deployment site. Shipments of fissile material are not uncommon, and there are several currently certified Type B packages suitable for this application. However, none that I could find are currently certified for the fissile material in the form of a fuel/salt mixture. This is because one of the accident scenarios involves a sustained fire at 1475 degrees Fahrenheit. Since FLiNaK melts at 850 degrees Fahrenheit, the fuel becomes mobile and potentially non-homogenous, and therefore significantly complicates the associated criticality analysis.

We will therefore have to design criticality safe fuel/salt transport containers which will fit inside the selected Type B container and ensure the mixture remains subcritical even as a non-homogenous, optimally reflected and moderated liquid. The simplest approach to accomplish this is to either use containers that are a criticality safe geometry, the incorporation of neutron poisons into the containers, or a combination of both. This type of container will also be required to transport the spent fuel/salt mixture, and will need to interface with the reprocessing facility equipment. All three of these operations should be incorporated into the final container design to construct a single container suitable for all three operations, thus providing maximum flexibility and minimum cost.

After reviewing the currently available over-the-road Type B transport packages, I selected the NAC-LWT, which is a shielded Type B cylindrical cask mounted inside a modified cargo container, for three reasons. First, it has a large internal cavity, 0.34 meters in diameter and 4.52 meters in length, with a capacity of 1,814 kg and a volume of $0.41m^3$. This geometry will support the criticality safe design (long and narrow) of the fuel transport containers. Second, it has a significant amount of lead shielding at just under 6 inches (14.6 cm). Third, because the package design conducts heat well, it has a decay heat limit of 1,260 watts.

The MSnB contains 3,187 kg of fuel/salt, with a corresponding volume of $0.932m^3$. Between the two, the volume is more restrictive, and requires 2.27 shipments. This leaves both adequate volume and weight for the crit-safe transport containers, meaning the MSnB fuel will require 3 shipments to transport the entire inventory to and from a deployment location.

To determine if the shielding is sufficient, I applied the gamma attenuation formula:

$$\chi = B\chi_0 e^{(-\mu/\rho)\rho x}$$

where χ is the shielded dose rate, χ_0 is the unshielded dose rate, μ/ρ is the mass atten-

Figure 5.5: NAC LWT fully assembled

Figure 5.6: NAC LWT cask only

uation coefficient for lead which varies as a function of photon energy, ρ is the density of lead, B is the buildup factor necessary to account for scatter within the shield material, and x is the required shield thickness.

I want to keep the shielded dose rate under 0.2 R/hr to simplify transportation compliance. I assumed an even division of fuel between the shipments, so the unshielded dose rate became 40,000/3 or 13,500 R/hr. I used a mass attenuation coefficient of $0.0976 cm^2/g$ which I obtained from the Health Physics Handbook based on a gamma energy of 750 keV, a density of $11.35 g/cm^3$ for the lead and a thickness of 14.6 cm. Also per the handbook, the build up factor is a function of the mass attenuation coefficient, the thickness of the required shielding, the energy of the gamma, and the density of the shield. For this scenario, the buildup is about 3.16. Therefore, the dose rate on the outside of the shielded cask will be

$$\chi = (3.16)(13500)e^{(-(0.0976)(14.6)(11.35))} = 4mr/hr$$

well within the 200 mr/hr limit.

Similarly, the decay heat associated with each shipment will be

$$DH = 1548/3 = 516W$$

well within the 1280 W limit.

5.3 External Shield Transport and Disposal

There are a few options concerning the construction of the external shield. If possible, the soil/rock at the deployment location would be used, eliminating the need to transport the shield components, as well as minimizing the size of the excavation required. However, the isotopic components of the soil/rock would have to be evaluated for activation products.

The neutron flux is still quite high immediately outside the reactor vessel during operation, in the range of $1(10)^{11} n/cm^2 s$. Another option would be to use a neutron absorbing shield, such as boron carbide, to eliminate activation of the surrounding soil/rock, while still allowing the soil/rock to shield the gamma dose. This option reduces both the amount of excavation necessary and the amount of shielding requiring transport. Another option is to transport all of the shielding necessary, and not rely on the local soil/rock at all. This option maximizes the amount of transport required, as well as maximizing the excavation if the reactor is placed underground. However, it might eliminate the need to excavate at all. I will use an intermediate approach, using ordinary concrete to shield the neutrons, and the surrounding soil/rock to shield the remaining gamma.

To estimate the gamma dose at contact with the outside of the reactor vessel during operation, I used the gamma exposure rate constant for each of the 187 tracked fission products, multiplied that against the activity of each fission product at the moment of reactor shutdown, divided by the square of the distance to the outer vessel surface (0.75 m), and then summed all of the resulting values. The result was approximately 3.4 million R/hr. This value only accounts the the gamma, and does not address the neutrons.

Using the assumption that 5 cm of concrete constitutes a Half Value Layer (HVL), about 26 HVLs would be required to reduce the dose rate to less than 100 mr/hr. Accordingly, that equates to a concrete shield about 130 cm (52 inches) thick surrounding the entire reactor vessel. The weight of such a shield would be approximately 125 tons.

To determine the point where activation of the surrounding earth is no longer a concern, I modified the MCNP model to add the concrete shield, and then a 1 cm thick concrete cylinder around that. I then adjusted the size of the concrete shield until the tally of neutron flux in the outer most thin shell dropped to zero. This occurred where the concrete shield had a thickness of 90 cm, with a corresponding mass of $7.6(10)^7 g$, or about 84 tons.

A standard commercial semi tractor trailer arrangement is capable of carrying 25 tons per load (40 ton road limit, with the tractor and trailer accounting for 15 tons). Accordingly, four loads would be sufficient to transport the entire shield. As such, it would have to be made of interlocking pieces, but could be cast as 2 cylinders, a bottom, and a lid. This simplifies both installation and removal, and provides an additional layer of defense to the environment by designing the bottom section to serve as a catch basin to any potential leaks of the molten salt. Assuming even weight distribution of the shield components, each component will weigh approximately $1.9(10)^7 g$, or 21 tons.

5.3.1 Shield Transportation

Assuming such leaks do not occur, the shield will not be radiologically contaminated, but it will contain the activation products identified previously. In order to use the LSA category III designation to ship the shield components, the following criteria must be satisfied:

1. The radioactive material is distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic, etc.);
2. The radioactive material is relatively insoluble, or it is intrinsically contained in a relatively insoluble material, so that even under loss of packaging, the loss of radioactive material per package by leaching, when placed in water for 7 days will not exceed 0.1 A2;
3. The estimated average specific activity of the solid, excluding any additional transport shielding material, does not exceed $2(10)^{-3}$ A2 per gram of waste;
4. The external dose rate may not exceed an external radiation level of 10 mSv/h (1 rem/h) at 3 m from the unshielded material;
5. The external dose rate on contact anywhere with the package exterior cannot exceed 200 mrem/hr, unless the material is shipped as an Exclusive Use shipment, in which case the limit becomes 1 Rem/hr.

Table 5.1 presents the relevant numbers to evaluate against the shipping criteria. As an

activation product	Activity (Ci)	A2 value (Ci)	(0.1)A2 (Ci)	$2(10)^{-3}$A2 (Ci/g)	$2(10)^{-3}$A2 (Ci/component)
H3	1.14E-2	20	2	0.02	$3.8(10)^5$
C14	4.19E-5	60	6	0.06	$1.14(10)^7$
K40	4.07E-5	24	2.4	.024	$3.8(10)^5$
Ca41	3.36E-1	unlimited			
Fe55	5.3E2	1100	110	1.1	$2.09(10)^7$

Table 5.1: Shield shipment summary

activation product, the radioactive materials are distributed throughout a solid, although not uniformly. The concentration of the activation products will be highest on the inside surfaces of the shield, and decrease substantially to the outside surfaces. The concrete is insoluble, the specific activity is of all activation products are quite low, and none of the activation products are gamma emitters. As such, the shield components qualify to ship under the LSA category III designation.

5.3.2 Shield Disposal

In order to dispose of the shielding components at NNSS, the shielding components will need to comply with two main source term requirements within the NNSS WAC:

1. The overall activity of the disposal package is controlled by converting all isotopic activity to an equivalent amount of Plutonium mass and then limiting the mass to 300 grams. The conversion factors are provided in Appendix B of the NNSS WAC;

2. Action levels are defined for isotopes which are particularly important to the performance assessment of the facility. If the action level is exceeded for any of these isotopes, then a re-evaluation of the repository performance with respect to the disposal package is required to determine if the package is acceptable for disposal. The action limits are provided in Appendix E of the NNSS WAC.

Table 5.2 summarizes the relevant transportation data. The low curie content ensures none of the limits are remotely close. As such, the shielding components easily qualify for disposal at the NNSS as low level waste.

activation product	Activity (Ci)	NNSS Pu-Equiv (PE-g/Ci)	NNSS action level (Ci/m^3)
H3	1.14E-2	2.67E-4	16.75
C14	4.19E-5	5.8E-3	1.5E5
K40	4.07E-5	0.0862	2.54
Ca41	3.36E-1	1.8E-4	75.7
Fe55	5.3E2	8E-4	none

Table 5.2: Shield disposal summary

5.4 Spent Reactor Vessel Transport and Disposal

The radiological source term remaining in the reactor vessel after the fuel/salt has been drained consists of three components; the activation products, the remnants of the fuel that remain after draining, and any non-soluble flourides formed with the fission products that adhere to the walls of the flow channels. Each of these is important to both transportation and disposal.

5.4.1 Void space

Of primary importance to any near surface burial site, void spaces within the waste materials are required to be eliminated to the greatest extent practical. This is because such voids will eventually collapse, resulting in subsidence of the soil surface directly above the waste. This forms a basin to collect water, which can then percolate down through the waste, mobilize the radioactive material, and result in contamination of the environment. The MSnB will have significant voids from the removal of the FLiNaK. Filling these voids with a stable grout, such as barite (high density) concrete will not only eliminate the void, but will also physically secure the radioactive particulate matter in place, allowing us to take advantage of the LSA shipment category, provide shielding of the internally activated components, and add strength and rigidity to the vessel. The weight of the concrete filled reactor vessel will remain around 32,000 pounds, as the density of barite concrete is about the same as the fuel/salt mixture (3.35 vs. 3.44 g/cm3).

5.4.2 Liquid remnant

Anytime a liquid is drained from a vessel, surface tension of the liquid results in some adhesion of the liquid to the wall of the vessel, preventing absolute removal of the liquid. The RCRA statutes have codified the definition of an empty vessel as containing no more than 3% of the liquid remaining after standard industrial operations of pouring or pumping have been performed (40 CFR 261). This is actually a fairly lenient standard, as it was designed to allow empty containers to be exempted from the hazardous waste requirements where a reasonable attempt had been made to remove the hazardous contents. 3% is a bounding value, and is based on the practical aspects of differing viscosities, liquid traps within the container, container geometries which prevent complete drainage, and the remnant left when pumping liquids from a container. A smooth walled container with a low viscosity liquid, such as a glass of water, will be drained to about 0.25% of its original volume just through the act of pouring. If the glass is left in the inverted position, even this can be reduced significantly.

The MSnB has a relatively simple geometry that should allow for good drainage. However, corrosion of the surfaces in contact with the molten salt will create pitting on those surfaces. Such pitting will both increase the overall surface area and create liquid traps. The actual amount of the fuel/salt that will remain is therefore probably somewhere between 0.25 and 3.0%.

The 3% remaining source term is summarized in Table 5.3. The NNSS action levels are determined on a Ci/m^3 basis, so I converted to curies based on the reactor vessel volume of 4.627 m3.

The large dose rate could be a problem depending on the distribution of the source within the vessel. Most of the surface areas exposed to the molten salt are near the outside portions of the reactor vessel, and therefore will not be adequately shielded. The limit for Pu-gram equivalence in a single package at NNSS is 300, so this portion of the

Isotope	Half-Life (yrs)	Activity (Ci)	LSA shipment Limit (Ci)	NNSS Action Level (Ci)	NNSS PE-g	Dose at 1 meter (R/hr)
Sr-89	0.14	814	2.90E+05	None	7	0
Sr-90	28.79	320	1.16E+04	53.67	51	0
Y-90	0.007	0.6	2.90E+05	None	0	0
Y-91	0.16	1,060	8.70E+06	None	10	2
Zr-95	0.18	1,219	5.80E+05	None	7	511
Nb-93m	16.12	544	5.80E+06	5.74E+05	1	0
Nb-95	0.1	927	5.80E+05	None	0	402
Mo-99	0.18	0.8	5.80E+05	None	0	0
Tc-99	2.11E+05	0.052	7.30E+05	0.4	0	0
Ru-103	0.11	463	7.30E+05	None	1	131
Te-132	0.009	1.7	2.00E+05	None	0	0
I-131	0.022	56	2.90E+04	None	0	12
Xe-133	0.014	33	2.90E+07	None	0	0.4
Cs-137	30	330	2.90E+05	31.27	13	106
Ba-140	0.035	315	5.80E+05	None	2	31.8
Ce-141	0.09	801	7.30E+05	None	3	0
Ce-144	0.78	1,326	2.00E+05	None	71	0
Pr-143	0.037	333	5.80E+05	None	1	0
Nd-147	0.03	88	5.80E+05	None	0	0
Pm-147	2.62	530	7.30E+05	None	4	0
Total					172	1197

Table 5.3: 3% remaining inventory

source term accounts for over half of the allowed quantity for the entire reactor vessel. The amounts of Sr-90 and Cs-137 are both problematic, as they exceed the NNSS action levels by a factor of 6.5 and 11.5 respectively. This means NNSS will have to reevaluate the performance of the repository with respect to this much activity, or more likely, we will need to reduce the inventory remaining in the vessel.

The 0.25% remaining inventory is summarized in Table 5.4. Both the dose rate and plutonium equivalent grams have been reduced by a factor of 12, making them far more manageable, and the action level issues have been eliminated. Accordingly, the MSnB design will have to incorporate features that allow at least this much drainage to occur. These features include appropriately sloped surfaces, elimination of liquid traps, minimization of corrosion to maintain smooth surfaces, and perhaps some type of rinsing operation. This might be accomplished by longer drain times or by rinsing the vessel after draining the fuel/salt mixture. In either case, this will need to be determined by testing.

Isotope	Half-Life (yrs)	Activity (Ci)	LSA shipment limit (Ci)	NNSS Action Level (Ci)	NNSS PE-g	Dose at 1 meter (R/hr)
Sr-89	0.14	68	2.90E+05	None	1	0
Sr-90	28.79	27	1.16E+04	53.67	4	0
Y-90	0.007	0.05	2.90E+05	None	0	0.018
Y-91	0.16	83	8.70E+06	None	1	0.168
Zr-95	0.18	102	5.80E+05	None	1	42.6
Nb-93m	16.12	45.3	5.80E+06	5.74E+05	0	0
Nb-95	0.1	77.25	5.80E+05	None	0	33.5
Mo-99	0.18	0.0667	5.80E+05	None	0	0
Tc-99	2.11E+05	0.0043	7.30E+05	0.4	0	0
Ru-103	0.11	38.6	7.30E+05	None	0	10.9
Te-132	0.009	0.1417	2.00E+05	None	0	0
I-131	0.022	4.67	2.90E+04	None	0	1.01
Xe-133	0.014	2.75	2.90E+07	None	0	0.0374
Cs-137	30	27.5	2.90E+05	31.27	1	8.86
Ba-140	0.035	26.25	5.80E+05	None	0	2.65
Ce-141	0.09	66.75	7.30E+05	None	0	0
Ce-144	0.78	111	2.00E+05	None	6	0
Pr-143	0.037	27.75	5.80E+05	None	0.07	0
Nd-147	0.03	7.33	5.80E+05	None	0	0
Pm-147	2.62	44.17	7.30E+05	None	0	0
Total					14	100

Table 5.4: 0.25% remaining inventory

5.4.3 Non-soluble fluorides

Some of the fission products, specifically the noble and semi-noble metals, form fluorides which are insoluble in the molten salt, and have a tendency to adhere to the flow channel surfaces. This issue was first observed during the Molten Salt Reactor Experiment (MSRE) in the 1960s, and is applicable to a significant number of fission products, including Zn, Ga, Ge, As, Se, Nb, Mo, Ru, Rh, Pd, Ag, Tc, Cd, In, Sn, Sb, Te, and I.

Because the MSRE continually processed the fuel salt mixture to remove fission poisons, and no other large scale molten salt reactors have operated since then, I found no information concerning the rate of deposition of these fission products onto reactor surfaces.

If all of these fission products adhered to the walls, the amounts would be as shown in Table 5.5. This is unacceptable for two reasons. First, because even with a $4.6m^3$ vessel, the NNSS action level for Tc99 is exceeded by over a factor of 4. Because of the long half life of Tc99, combined with its high mobility, it will be difficult for NNSS to justify the exception. Second, the extreme gamma dose rate will complicate shipment. If we limit the inventory of these fission products to 20%, the action level problem is eliminated, and the dose reduced by a factor of 5 to 3,634 R/hr. Although the intent is to be able to ship within 30 days of reactor shutdown, this area may be an exception. All of the dose from this inventory is associated with short-lived fission products, which may decay to acceptable shipment levels within a matter of weeks.

Accordingly, the MSnB design must limit the deposition of this material, particularly the Tc99, to no more than 20% of the total inventory. Empirical evaluations will need to be performed to quantify the rate of deposition on various surfaces and various levels of corrosion. If the results indicate more than 20% will adhere, then additional design features such as a chemical rinse to remove the source term after the fuel has drained, or periodic processing of the salt to remove the source term.

5.4.4 Activation Products

Table 5.6 provides a summary of the vessel activation products and associated limits. Co-60 exceeds the NNSS action limit by a factor of about 60, and by itself exceeds the allowable Pu-gram equivalence. Accordingly, the MSnB design will need to significantly reduce the amount of Co-60.

The best alternative would be to replace the steel and hastelloy components within the reactor vessel with alternative materials that do not contain either cobalt or molyb-

Isotope	Half-Life (yrs)	Activity Limit (Ci)	LSA ship Limit (Ci)	NNSS Act Level (Ci/m3)	NNSS PE-g	Dose at 1 Meter (R/hr)
Nb-93m	16.12	1.81E+04	5.80E+06	1.24E+05	33	0
Nb-95	0.1	3.09E+04	5.80E+05	None	5	13406
Mo-99	0.18	27	5.80E+05	None	1	2
Tc-99	2.11E+05	1.73	7.30E+05	0.0865	0	0
Ru-103	0.11	1.54E+04	7.30E+05	None	46	4350
Te-132	0.009	56.7	2.00E+05	None	0	7
I-131	0.022	1.87E+03	2.90E+04	None	14	404
I-133	0.00238	8.40E+04	4.65E+05	None	0	0
Totals					99	18169

Table 5.5: Entire Non-Soluble Fission Inventory

Isotope	Half-Life (yrs)	Activity (Ci)	LSA shipment Limit (Ci)	NNSS Action Level (Ci)	NNSS Pu-g Equivalence
Co-60	5.3	1.20E+04	3.20E+05	1.99E+02	3.76E+02
Fe-55	2.7	7.51E+04	3.20E+07	None	5.97E+01
Ni-59	76,000	3.65E+01	unlimited	2.11E+04	1.65E-02
Ni-63	101.2	3.93E+03	2.36E+07	3.98E+04	5.03E+00
Tc-99	211,000	7.88E-02	6.98E+05	3.98E-01	1.07E-03
Be-10	1.50E+06	5.38E-01	4.65E+05	None	1.89E-02
Total					441

Table 5.6: Vessel Activation summary

denum, eliminating both the Co60 and the Tc99. This involves an extensive research effort, as the replacement materials must withstand the neutron bombardment and exposure to 700 degree C molten FLiNaK, without excessive deformation, and still provide adequate strength and corrosion resistance. Nonetheless, research has been initiated on many types of ceramics which may be suitable, including Silicon Carbide, Silicon Nitride, and Aluminum Nitride.

Almost 60% of the cobalt is generated in the 1 cm thick inner vessel, where the neutron flux is at its highest value outside the core. The purpose of the inner vessel is to provide the flow channel for the molten salt. Investigations will need to be performed to determine if it is feasible to eliminate this surface, and simply allow the beryllium oxide to contact the molten salt directly. If this is not feasible, then perhaps the inner vessel can be replaced with a graphite vessel.

Components further from the core such as the hastelloy pivot pins could be shielded to some degree with boron carbide, reducing the flux and therefore the quantity of activation products. The outer vessel could be similarly shielded, or perhaps reduced in thickness.

In any case, the Co60 does not have to be eliminated entirely. Assuming we can eliminate the inner vessel, the Co60 curie content drops to 4785 curies. This source term is not a point source, but is distributed in a non-homogenous manner throughout the reactor vessel, including the outer vessel itself. Nonetheless, assuming it is a point source will provide an approximation of the shielding required.

The dose rate at contact with the vessel outer surface from this source becomes

$$DR = A_x \Gamma_x / d^2 = 1.29(4785)/.75^2 = 1.105(10)^4 R/hr$$

Since the reactor void spaces have been filled with barite concrete, there is significant self-shielding. The HVL for ordinary concrete associated with Co-60 is 6.2 cm. Ordinary concrete is similar in density to both the boron carbide and the beryllium oxide. The

HVL for barite concrete is 4 cm. Therefore, 5 cm is considered a reasonable estimate of the HVL for the combined reactor materials, and this provides around 15 HVLs of self-shielding, reducing the exposure rate on the external surface of the outer reactor vessel to

$$DR = 11048/2^{15} = 0.534 Rem/hr = 337 mrem/hr$$

Shipment as LSA category III requires the reactor vessel to either be enclosed within an IP-3 container for standard transport, with an associated dose rate of less than 200 mrem/hr on contact anywhere on the exterior of the container, or an IP-2 container for exclusive use transport, with an associated dose rate of less than 1 Rem/hr. Standard intermodel shipping containers (Figure 5.7) are available as IP-2, IP-3, and even Type A compliant. The MSnB will need one specifically constructed for the vessel, as the standard sizes would not quite accommodate the physical dimensions of the MSnB.

The reactor vessel is then placed into the intermodal container, and the void space again filled with grout. This is done to comply with the void space elimination requirement at NNSS, secure the vessel inside the container for transport, and allow the entire intermodal to be disposed, eliminating exposure of personnel at the disposal site.

Since the reactor vessel weighs approximately 32,000 pounds, and the tare weight of an intermodal is about 4,000 pounds, approximately 12,000 pounds of grout could be used. The container design would be adjusted to provide maximum shielding while ensuring compliance with the transport weight limits. An IP-3 compliant container would probably suffice, as it would provide additional distance and shielding and probably meet the 200 mrem/hr limit, even with a higher dose rate from a distributed source. If not, an IP-2 compliant container would work in conjunction with an exclusive use shipment designation.

This approach does not eliminate the problem with Co60 exceeding the NNSS action limit. 4940 curies will still exceed the limit by a factor of 25. However, since the Co60 is

Figure 5.7: Intermodal shipping container

an activation product, and therefore locked into the matrix of the 304 and the Hastelloy, migration of this radiological material to the environment requires the degradation of the steel. The Co60 will almost entirely decay away within 50 years (10 half-lives), long before the steel has a chance to significantly corrode. For this reason, the repository performance assessment review required by NNSS may not be significant.

CHAPTER 6

Summary and Conclusions

In this thesis, I have performed a detailed preliminary transportation and disposal analysis of a 400 kW(th) molten salt reactor operating for 10 years. This analysis has shown that it is feasible to design and construct the reactor such that it can be truly mobile, meaning it can be deployed and recovered within weeks of reactor shutdown using only standard commercially available over-the-road transport equipment. Further, all components other than the spent fuel can be immediately disposed as LLW. All of this can be done without special permits and exemptions from the existing regulatory framework.

The design changes necessary to ensure this outcome consist of the following:

1. Modularize the design such that no single component is greater than 50,000 pounds, and each component is transportable on a standard commercial semi tractor/trailer assembly;

2. Design criticality safe transport containers for the fuel/salt mixture that will fit inside the NAC LWT shipping cask, and ensure the fuel/salt mixture remains subcritical even in the most optimally moderated and reflected geometry;

3. Design the reactor to allow remote operations to drain the fuel into the transport containers, as well as allowing remote operations to disconnect the containers from the reactor vessel;

4. Design the reactor to allow complete filling of the void spaces in the reactor with high density grout once the fuel/salt has been drained;

5. Eliminate liquid traps and slope all surfaces to ensure that no more than 0.25% of the fuel/salt inventory remains after drainage. Incorporate a rinsing system if necessary;

6. Investigate and if necessary, incorporate mechanisms to keep the non-soluble fluo-

rides from depositing as scale on the internal surfaces of the reactor vessel to a level greater than 20% of the inventory, either by controlling corrosion, or incorporating a chemical rinse system;

7. Eliminate or at least minimize the amount of Co60 and Tc99 by replacing the stainless steel and hastelloy components with ceramics made of silica or alumina, or with graphite.

CHAPTER 7

Future Work

Four significant areas of additional investigation are required to support the design changes identified in the previous chapter, as outlined below.

The primary effort will be evaluating alternative materials for the 304 stainless and the Hastelloy steels for the purpose of eliminating the associated activation products, particularly the Co60 and Tc99. The alternative materials will need to possess adequate strength, corrosion resistance, temperature stability, and tolerance of neutron bombardment relative to limited distortion, stress cracking and embrittlement. Because the MSnB is not a pressurized reactor, the ductile properties of the steels may not be required, and therefore ceramics are probably the most promising alternatives. Part of this effort should be an evaluation to simply eliminate some of the internal steel components. If the reflector material can withstand direct contact with the molten salt, there may be no need for the inner vessel or chimney components.

The next most important effort will involve the empirical studies needed to quantify the reactor vessel radioactive source term from both the liquid remnant which remains after drainage, as well as the deposition of the insoluble fluorides. These studies will need to be performed at reactor operating temperatures, and will be dependent upon the amount of corrosion expected to occur during the life of the reactor, as the added surface roughness will affect both drainage and scale adhesion. Additionally, if these studies indicate either source term component will be a problem, potential remedies (e.g., rinsing) will need to be evaluated.

The third effort will be the design of the spent fuel/salt mixture transport containers. These will need to be tested (or at least modeled), and certified by the NRC as an acceptable inner container for the NAC-LWT cask to ensure the mixture remains subcritical during all postulated accident scenarios. This design will need to address several factors, including:

a) How the containers will be positioned under the reactor vessel to ensure filling by gravity drainage of the vessel;

b) How the containers will be connected to the vessel to allow for separation of the two remotely, and without the need for additional radiological contamination confinement systems;

c) How the drainage valves will be configured and operated to ensure each container is filled only to the appropriate level;

d) How the individual containers will be transferred to the NAC-LWT (i.e., use of a shielded transfer cask vs. additional shielding incorporated into the design of the individual containers);

e) How the containers will be heated to allow draining to reprocess the spent fuel.

The fourth effort will involve the design details for the external shield. The design will need to ensure:

a) the individual pieces can be easily assembled and disassembled remotely;

b) adequate shielding of the spent fuel/salt transport containers;

c) potential spill containment and recovery;

d) adequate top side penetrations needed to access connections with the heat exchanger, fuel/salt fill port, and any monitoring and control cabling.

References

[11] Juan A. Vitalie et. al, US Army Study on the use of Mobile Nuclear Power Plants (2018)

[4] Lawrence H. Suid, The Army's Nuclear Power Program, published in Contributions in Miliary Studies, Number 98, 1990

[2] Stacy, Susan M., Proving the Principle, ISBN 0-16-059185-6, 2000

[5] Historic Fort Belvoir SM-1 Reactor to be Decommissioned, published in Nuclear Newswire, September 1, 2020

[3] Wendorf, Marcia, U.S.S. Thresher and Scorpion - The U.S.'s Lost Nuclear Submarines, published in Interesting Engineering Innovation April 14, 2019

[1] Corliss, William R., Power Reactors in Small Packages, US AEC Technical Information Pamphlet June 1964

[7] US Army Final Report on Construction of the Army Nuclear Power Plant PM-2A at Camp Century Greenland, August 1961

[9] Reid, Tyler, Nuclear Power at McMurdo Station, Antarctica, published by Stanford University, 2014

[10] The ML-1 Design Report, Aerojet-General May 1960

[6] Nuclear Engineering International newsletter, July 8, 2021

[8] Clark, Elmer F., Camp Century Evolution of Concept and History of Design Construction and Performance, US Army Materiel Command Technical Report 174, October 1965

[12] Carter, John et. al, Microscale Nuclear Battery Shielding and Fuel Cycle, University of Idaho Advanced Reactor Design Class Project Report, Spring 2021

Appendix A: MCNP Model of MSnB

This Appendix provides the MCNP card deck used to create the reactor model:

```
MSnB reactor model by Dave Haar
C Cell cards
1 1 -3.02 -1 2 3 7 11 15 19 23 27 31 imp:n=1 $ reflector
2 2 -3.3869 -41 imp:n=1 $ core
3 3 -2.52 -3 4 5 6 imp:n=1 $ control drum 1
4 1 -3.02 -3 4 -5 6 imp:n=1 $ reflector portion of drum 1
5 1 -3.02 -3 -4 5 6 imp:n=1 $ reflector portion of drum 1
6 1 -3.02 -3 -4 -5 6 imp:n=1 $ reflector portion of drum 1
7 4 -8.86 -6 imp:n=1 $ hastelloy pivot rod for 1
8 3 -2.52 -7 8 9 10 imp:n=1 $ control drum 5
9 1 -3.02 -7 8 -9 10 imp:n=1 $ reflector portion of drum 5
10 1 -3.02 -7 -8 9 10 imp:n=1 $ reflector portion of drum 5
11 1 -3.02 -7 -8 -9 10 imp:n=1 $ reflector portion of drum 5
12 4 -8.86 -10 imp:n=1 $ hastelloy pivot rod for 5
13 3 -2.52 -11 12 13 14 imp:n=1 $ control drum 3
14 1 -3.02 -11 12 -13 14 imp:n=1 $ reflector portion of drum 3
15 1 -3.02 -11 -12 13 14 imp:n=1 $ reflector portion of drum 3
16 1 -3.02 -11 -12 -13 14 imp:n=1 $ reflector portion of drum 3
17 4 -8.86 -14 imp:n=1 $ hastelloy pivot rod for 3
18 3 -2.52 -15 16 17 18 imp:n=1 $ control drum 7
19 1 -3.02 -15 16 -17 18 imp:n=1 $ reflector portion of drum 7
20 1 -3.02 -15 -16 17 18 imp:n=1 $ reflector portion of drum 7
21 1 -3.02 -15 -16 -17 18 imp:n=1 $ reflector portion of drum 7
```

```
22 4 -8.86 -18 imp:n=1 $ hastelloy pivot rod for 7
23 3 -2.52 -19 20 21 22 imp:n=1 $ control drum 2
24 1 -3.02 -19 20 -21 22 imp:n=1 $ reflector portion of drum 2
25 1 -3.02 -19 -20 21 22 imp:n=1 $ reflector portion of drum 2
26 1 -3.02 -19 -20 -21 22 imp:n=1 $ reflector portion of drum 2
27 4 -8.86 -22 imp:n=1 $ hastelloy pivot rod for 2
28 3 -2.52 -23 24 25 26 imp:n=1 $ control drum 6
29 1 -3.02 -23 24 -25 26 imp:n=1 $ reflector portion of drum 6
30 1 -3.02 -23 -24 25 26 imp:n=1 $ reflector portion of drum 6
31 1 -3.02 -23 -24 -25 26 imp:n=1 $ reflector portion of drum 6
32 4 -8.86 -26 imp:n=1 $ hastelloy pivot rod for 6
33 3 -2.52 -27 28 29 30 imp:n=1 $ control drum 4
34 1 -3.02 -27 28 -29 30 imp:n=1 $ reflector portion of drum 4
35 1 -3.02 -27 -28 29 30 imp:n=1 $ reflector portion of drum 4
36 1 -3.02 -27 -28 -29 30 imp:n=1 $ reflector portion of drum 4
37 4 -8.86 -30 imp:n=1 $ hastelloy pivot rod for 4
38 3 -2.52 -31 32 33 34 imp:n=1 $ control drum 8
39 1 -3.02 -31 32 -33 34 imp:n=1 $ reflector portion of drum 8
40 1 -3.02 -31 -32 33 34 imp:n=1 $ reflector portion of drum 8
41 1 -3.02 -31 -32 -33 34 imp:n=1 $ reflector portion of drum 8
42 4 -8.86 -34 imp:n=1 $ hastelloy pivot rod for 8
43 5 -7.95 -35 36 imp:n=1 $ 304 SS outer reactor vessel
44 6 -3.3869 -36 1 6 10 14 18 22 26 30 34 37 38 40 42 43 44 imp:n=1 $ fuel
45 3 -2.52 -37 imp:n=1 $ bottom absorber plate
46 5 -7.95 -38 imp:n=1 $ bottom steel plate just above absorber
47 5 -7.95 -40 39 imp:n=1 $ chimney 304 SS wall
48 6 -3.3869 -39 imp:n=1 $ fuel inside the chimney
```

```
49 5 -7.95 -2 41 imp:n=1 $ inner reactor vessel 304 SS wall
50 3 -2.52 -42 40 imp:n=1 $ upper absorber around chimney
51 7 -0.001205 -43 6 10 14 18 22 26 30 34 imp:n=1 $ air chamber
52 8 -2.6989 -44 45 imp:n=1 $ aluminum portion of heat exchanger
53 6 -3.3869 -45 6 10 14 18 22 26 30 34 imp:n=1 $ fuel inside aluminum cylinder
54 9 -2.3 -46 35 imp:n=1 $ external concrete shield
55 9 -2.3 -47 46 imp:n=1 $ determination of neutron flux outside reactor
56 0 47 imp:n=0
C Surface Cards
1 rcc 0 0 0 0 0 167 64 $reflector module outer surface
2 rcc 0 0 0 0 0 167 26 $reflector module inner surface
3 rcc 0 41.5 0 0 0 167 15 $control drum 1
4 p -1 1.39 0 57.6781 $ intersecting plane in drum 1
5 p 1 1.39 0 57.6781 $ other intersecting plane in drum 1
6 rcc 0 41.5 0 0 0 225 2.5 $ hastelloy pivot rod 1
7 rcc 0 -41.5 0 0 0 167 15 $ control drum 5
8 p -1 -1.39 0 57.6781 $ intersecting plane in drum 5
9 p 1 -1.39 0 57.6781 $ other intersecting plane in drum 5
10 rcc 0 -41.5 0 0 0 225 2.5 $ hastelloy pivot rod 5
11 rcc 41.5 0 0 0 0 167 15 $ control drum 3
12 p 1.39 1 0 57.6781 $ intersecting plane in drum 3
13 p 1.39 -1 0 57.6781 $ other intersecting plane in drum 3
14 rcc 41.5 0 0 0 0 225 2.5 $ hastelloy pivot rod 3
15 rcc -41.5 0 0 0 0 167 15 $ control drum 7
16 p -1.39 1 0 57.6781 $ intersecting plane in drum 7
17 p -1.39 -1 0 57.6781 $ other intersecting plane in drum 7
18 rcc -41.5 0 0 0 0 225 2.5 $ hastelloy pivot rod 7
```

```
19 rcc 29.345 29.345 0 0 0 167 15 $ control drum 2
20 p 3.8 1 0 142.1 $ intersecting plane in drum 2
21 p 1 3.8 0 142.1 $ other intersecting plane in drum 2
22 rcc 29.345 29.345 0 0 0 225 2.5 $ hastelloy pivot rod 2
23 rcc -29.345 -29.345 0 0 0 167 15 $ control drum 6
24 p -3.8 -1 0 142.1 $ intersecting plane in drum 6
25 p -1 -3.8 0 142.1 $ other intersecting plane in drum 6
26 rcc -29.345 -29.345 0 0 0 225 2.5 $ hastelloy pivot rod 6
27 rcc 29.345 -29.345 0 0 0 167 15 $ control drum 4
28 p 3.8 -1 0 142.1 $ intersecting plane in drum 4
29 p 1 -3.8 0 142.1 $ other intersecting plane in drum 4
30 rcc 29.345 -29.345 0 0 0 225 2.5 $ hastelloy pivot rod 4
31 rcc -29.345 29.345 0 0 0 167 15 $ control drum 8
32 p -3.8 1 0 142.1 $ intersecting plane in drum 8
33 p -1 3.8 0 142.1 $ other intersecting plane in drum 8
34 rcc -29.345 29.345 0 0 0 225 2.5 $ hastelloy pivot rod 8
35 rcc 0 0 -26 0 0 255 76 $ outer surface of outer reactor vessel
36 rcc 0 0 -23 0 0 249 73 $ inner surface of outer reactor vessel
37 rcc 0 0 -23 0 0 10 73 $ bottom absorber plate
38 rcc 0 0 -13 0 0 1 73 $ bottom steel plate just above absorber
39 rcc 0 0 167 0 0 37 25 $ chimney inner wall surface
40 rcc 0 0 167 0 0 37 26 $chimney outer wall
41 rcc 0 0 0 0 0 167 25 $inner reactor vessel inner wall
42 rcc 0 0 167 0 0 37 28 $ upper absorber around chimney
43 rcc 0 0 215 0 0 11 73 $ air flow cylinder
44 rcc 0 0 209 0 0 6 73 $ aluminum outer cylinder of heat exchanger
45 rcc 0 0 209 0 0 6 45 $ aluminum clylinder inner surface
```

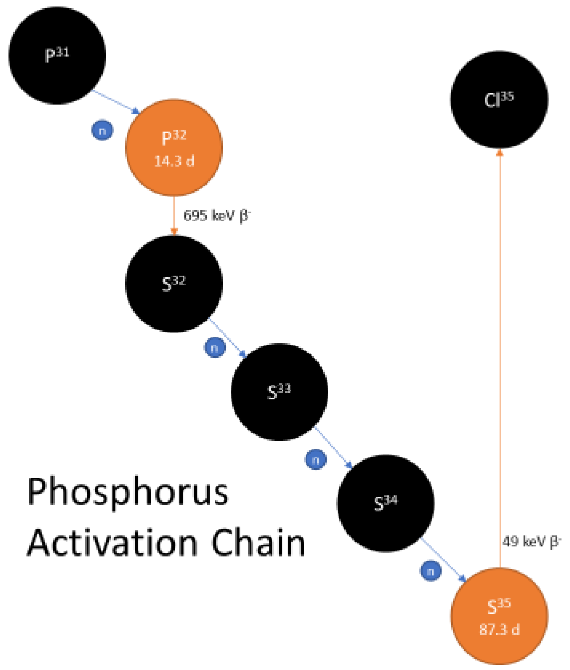
```
46 rcc 0 0 -116 0 0 435 166 $ external shield outer surface
47 rcc 0 0 -117 0 0 437 167 $ external concrete just to determine flux
C Data cards
C Criticality Control Cards
kcode 5000 1.0 50 250
ksrc 0 0 125
C Material cards
m1 4009 0.5 $ beryllium oxide
8016 0.5
m2 92235.66c 0.0234 $enriched to 32 percent
92238.66c 0.0498
3007.66c 0.1474
11023.66c 0.0365
19039 0.1242
19041 0.009
9019.66c 0.6097
m3 5011 0.799981 $ boron carbide
6012 0.200018
m4 26000 .05  hastelloy with impurities
24000 0.07 $ Cr
29063 0.0242 $ Cu63
29065 0.0106 $ Cu65
28000 0.6672 $ Ni
25055 0.0008 $ Mn
14000 0.001 $ Si
42000 0.16 $ Mo
6012 0.0008 $ C
```

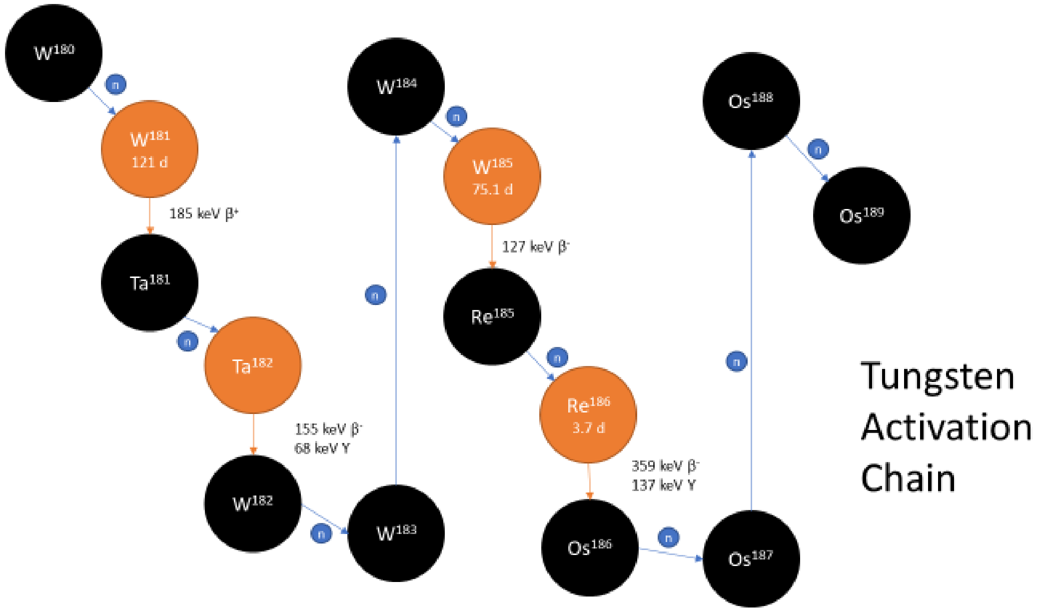
```
27059 0.002 $ Co
15031 0.00045 $ P
16032 0.0003 $ S
74000 0.005 $ W
13027 0.0018 $ Al
22000 0.0018 $ Ti
m5 26000 .69945 $ 304 SS with impurities
24052 0.1508
24053 0.0171
24054 0.0043
28000 0.08
25055 0.02
14000 0.0075
42000 0.003
6012 0.0008
27059 0.0005
15031 0.00045
16032 0.0003
41093 0.007
7014 0.001
m6 92235.66c 0.0145 92238.66c 0.0587
3007.66c 0.1474 11023.66c 0.0365
19039 0.1242 19041 0.009
9019.66c 0.6097
m7 6000 0.000151 $ air - first element C
7014 0.784437 $ N
8016 0.210750 $ O
```

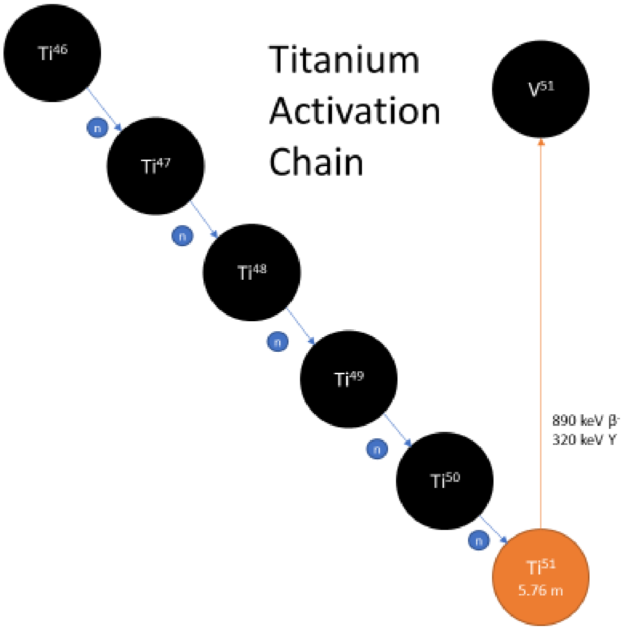
```
18000 0.004671 $ Ar
m8 13027 1.000000 $ Aluminum
m9 1001 0.304245 $H Concrete
6012 0.002870 $C
8016 0.498628 $O
11023 0.009179 $Na
12000 0.000717 $Mg
13027 0.010261 $Al
14000 0.150505 $Si
19000 0.007114 $K
20000 0.014882 $Ca
26000 0.001599 $Fe
C Tally Cards
f4:n 2 7 43 45 46 47 49 50 54 55
```

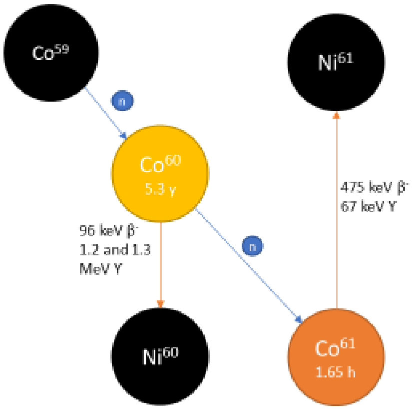
Appendix B: Activation Calculations

Appendix B.1. Activation Chains for Various Impurities






Titanium Activation Chain



Cobalt Activation Chain

Appendix B.2. Reactor Component Activation Calculations

This appendix provides the calculations associated with determining the activation products for the various reactor components. The formula used in these calculations is

$$A_x = N_x \lambda_x = \sigma N_p \phi (1 - e^{-\lambda_x t})$$

The flux was obtained from the MCNP model. The cross section for thermal neutron capture was obtained from the IAEA's live Atlas of Neutron Capture Cross Sections. The decay constant was obtained from the IAEA's live Chart of the Nuclides.

The total number of molecules in a given component was obtained by multiplying the volume of the component by its density and Avagadro's number, and dividing by the molecular weight:

$$N_{TM} = V N_A \rho / MW$$

Results are provided in the following table:

Component	volume (cm^3)	density (g/cm^3)	molecular weight ($g/mole$)	Total number of molecules
Inner 304 reactor vessel	5.35E+04	7.95	54.58	4.69E+27
304 chimney	1.19E+04	7.95	54.58	1.04E+27
304 outer vessel	4.59E+05	7.95	54.58	4.02E+28
304 Bottom plate	1.67E+04	7.95	54.58	1.47E+27
Hastelly pivot rods (all 8)	3.53E+04	8.86	54.58	3.46E+27
Beryllium Oxide	1.49E+06	3.2	12.5	2.29E+29
Boron Carbide in control drums	2.83E+05	2.52	11.32	3.80E+28
Boron Carbide in bottom absorber	1.67E+05	2.52	11.32	2.24E+28
Boron Carbide in upper absorber	1.26E+04	2.52	11.32	1.68E+27
Graphite in core	1.39E+04	2.27	12.01	1.58E+27

The methodology for the concrete in the shield was different, as I was not sure of the chemical formula. Instead, I used the Compendium of Material Composition Data for Radiation Transport Modeling published by the Pacific Northwest National Laboratory to identify the mass fractions of the various elements which make up ordinary concrete. For the elements of concern, I multiplied the concrete mass value ($7.6E7$ g) by the mass fraction to obtain the mass of each element. Then divided by the molecular weight to obtain the number of moles of each element, and finally multiplied by Avagadro's number to get the number of atoms of each element. The results are summarized below:

Isotope	Weight Fraction	Mass of Elemental (g)	Molecular Weight g/mole)	Moles of Element	Total Atoms of Element
H	0.0221	1.68E6	1.008	1.66E6	1.00E30
C	.002484	1.89E5	12.01	1.57E4	9.47E27
K	.010045	7.63E5	39.098	1.95E4	1.17E28
Ca	.042951	3.26E6	40.078	8.14E5	4.91E28
Fe	.006435	4.89E5	55.845	8.76E3	5.27E27

The total number of parent atoms in the component was then determined by multiplying this number by atom percentage and the abundance of the parent in the material, as shown in the following table:

Component	Total number of molecules	Activation Product	Parent Atom	Abundance % of Parent	Atom % of Parent	Number of Parent atoms
Inner Vessel	4.69E+27	Co-60	Co-59	100	0.05	2.35E+24
		Fe-55	Fe-54	5.84	69.945	1.92E+26
		Ni-59	Ni-58	68.3	8	2.57E+26
		Ni-63	Ni-62	3.6	8	1.35E+25
Chimney	1.04E+27	Co-60	Co-59	100	0.05	5.20E+23
		Fe-55	Fe-54	5.84	69.945	4.25E+25
		Ni-59	Ni-58	68.3	8	5.68E+25
		Ni-63	Ni-62	3.6	8	3.00E+24
Outer Vessel	4.02E+28	Co-60	Co-59	100	0.05	2.01E+25
		Fe-55	Fe-54	5.84	69.945	1.64E+27
		Ni-59	Ni-58	68.3	8	2.20E+27
		Ni-63	Ni-62	3.6	8	1.16E+26
Bottom plate	1.47E+27	Co-60	Co-59	100	0.05	7.34E+23
		Fe-55	Fe-54	5.84	69.945	6.00E+25
		Ni-59	Ni-58	68.3	8	8.03E+25
		Ni-63	Ni-62	3.6	8	4.23E+24
Hastelloy rods	3.46E+27	Co-60	Co-59	100	0.1	3.46E+24
		Fe-55	Fe-54	5.84	3	6.05E+24
		Ni-59	Ni-58	68.3	72.185	1.70E+27
		Ni-63	Ni-62	3.6	72.185	8.98E+25
		Mo-93	Mo-92	14.53	16.5	8.34E+25
		Nb-94	Nb-93	n/a	n/a	1.28E+17
		Tc-99	Mo-98	24.3	16.5	1.39E+26
BeO	2.29E+29	Be-10	Be-9	100	50	1.15E+29
B_4C drums	3.80E+28	C-14	C-13	1.1	24	1.00E+26
B_4C l abs	2.24E+28	C-14	C-13	1.1	24	5.93E+25
B_4C u abs	1.68E+27	C-14	C-13	1.1	24	4.44E+24
Graphite	1.58E+27	C14	C-13	1.1	100	1.74E+25

and for the concrete shield:

Element	Total Atoms of Element	Activation Product	Parent Atom	Abundance % of Parent	Number of Parent Atoms
H	1.00E30	H3	H2	0.0115	1.15E26
C	9.47E27	C14	C13	1.07	1.01E26
K	1.17E28	K40	K39	93.2581	1.09E28
Ca	4.91E28	Ca41	Ca40	96.94	4.76E28
Fe	5.27E27	Fe55	Fe54	5.845	3.08E26

Once the number of parent atoms have been determined, then the formula

$$A_x = N_x \lambda_x = \sigma N_p \phi (1 - e^{-\lambda_x t})$$

can be applied, with the results as shown in the following:

Component	Act Prod	Number Parent Atoms	Capture cross section (cm^2)	Flux (n/cm^2s)	Decay constant ($/s$)	Activity (Ci)
Inner Vessel	Co-60	2.35E+24	3.7E-23	3.97E+12	4.15E-09	6.80E+03
	Fe-55	1.96E+26	2.70E-24	3.97E+12	8.02E-09	5.11E+04
	Ni-59	2.57E+26	4.60E-24	3.97E+12	2.89E-13	1.15E+01
	Ni-63	1.35E+25	1.46E-23	3.97E+12	2.17E-10	1.40E+03
Chimney	Co-60	5.20E+23	3.70E-23	1.38E+12	4.15E-09	5.24E+02
	Fe-55	4.25E+25	2.70E-24	1.38E+12	8.02E-09	3.94E+03
	Ni-59	5.68E+25	4.60E-24	1.38E+12	2.89E-13	8.89E-01
	Ni-63	3.00E+24	1.46E-23	1.38E+12	2.17E-10	1.08E+02
Outer Vessel	Co-60	2.01E+25	3.70E-23	1.29E+11	4.15E-09	1.89E+03
	Fe-55	1.64E+27	2.70E-24	1.29E+11	8.02E-09	1.42E+04
	Ni-59	2.20E+27	4.60E-24	1.29E+11	2.89E-13	3.21E+00
	Ni-63	1.16E+26	1.46E-23	1.29E+11	2.17E-10	3.90E+02
Bottom Plate	Co-60	7.34E+23	3.70E-23	7.67E+11	4.15E-09	4.11E+02
	Fe-55	6.00E+25	2.70E-24	7.67E+11	8.02E-09	3.09E+03
	Ni-59	8.03E+25	4.60E-24	7.67E+11	2.89E-13	6.98E-01
	Ni-63	4.23E+24	1.46E-23	7.67E+11	2.17E-10	8.47E+01
Hastelloy Rods	Co-60	3.46E+24	3.70E-23	7.76E+11	4.15E-09	1.96E+03
	Fe-55	6.05E+24	2.70E-24	7.76E+11	8.02E-09	3.15E+02
	Ni-59	1.70E+27	4.60E-24	7.76E+11	2.89E-13	1.50E+01
	Ni-63	8.98E+25	1.46E-23	7.76E+11	2.17E-10	1.82E+03
	Mo-93	8.34E+25	2.ooE-26	7.76E+11	5.49E-12	1.10E-02
	Nb-94	1.28E+17	2.ooE-25	7.76E+11	1.08E-12	3.33E-11
	Tc-99	1.39E+26	8.00E-25	7.76E+11	1.04E-13	7.63E-02
BeO	Be-10	1.15E+29	8.00E-27	4.58E+12	1.46E-14	5.21E-01
B_4C-drums	C-14	1.00E+26	1.00E-27	7.50E+11	3.86E-12	2.47E-03
B_4C-upper abs	C-14	5.93E+25	1.00E-27	5.21E+11	3.86E-12	1.01E-03
B_4C-lower abs	C-14	4.44E+24	1.00E-27	1.33E+12	3.86E-12	1.94E-04
Graphite	C-14	1.74E+25	1.00E-27	4.21E+12	1.79E-09	8.52E-01
Concrete	H-3	1.15E26	1.00E-27	6.48E9	1.79E-9	8.69E-3
	C-14	1.01E26	1.00E-27	6.48E9	3.85E-12	2.15E-5
	K-40	1.09E28	2.00E-24	6.48E9	1.76E-17	2.13E-5
	Ca-41	4.76E28	5.00E-25	6.48E9	2.21E-13	2.9E-1
	Fe-55	3.08E26	2.00E-24	6.48E9	8.00E-9	9.93E1

Appendix C: Fission Product Calculations

Appendix C.1

This Appendix identifies the 187 fission products tracked by the Serpent code, and quantifies the activity of each one. The formula

$$A_x = F_s Y_x (1 - e^{-\lambda t})$$

was used to determine each activity at the end of the 10 year operating period. The formula

$$A_x = A_{x0}(e^{-\lambda t})$$

was then applied to determine the activity remaining at 30 days post shutdown.

Isotope	Cumulative Yield (/f)	Number of atoms	Half Life (s)	Decay Constant (/s)	Activity at end of life (Beq)	Activity at 30 days (Beq)
Se-86	1.37E-02	1.38E+23	1.43E+01	4.85E-02	4.37E+14	0
Br-85	1.28E-02	1.30E+23	1.74E+02	3.98E-03	4.11E+14	0
Br-86	1.59E-02	1.61E+23	5.50E+01	1.26E-02	5.10E+14	0
Br-87	2.03E-02	2.05E+23	5.60E+01	1.24E-02	6.51E+14	0
Br-88	1.78E-02	1.80E+23	1.60E+01	4.33E-02	5.69E+14	0
Br-89	1.09E-02	1.10E+23	4.00E+00	1.73E-01	3.47E+14	0
Kr-84m	1.00E-02	1.01E+23	1.83E-06	3.79E+05	3.21E+14	0
Kr-85m	1.29E-02	1.30E+23	1.61E+04	4.30E-05	4.13E+14	0
Kr-86	1.96E-02	1.98E+23	stable			
Kr-87	2.56E-02	2.58E+23	4.58E+03	1.51E-04	8.18E+14	0
Kr-88	3.55E-02	3.59E+23	1.02E+04	6.79E-05	1.14E+15	0
Kr-89	4.51E-02	4.56E+23	1.89E+02	3.67E-03	1.44E+15	0
Kr-90	4.86E-02	4.91E+23	3.20E+01	2.17E-02	1.56E+15	0
Kr-91	3.35E-02	3.38E+23	8.70E+00	7.97E-02	1.07E+15	0
Kr-92	1.67E-02	1.69E+23	2.00E+00	3.47E-01	5.36E+14	0
Rb-85	1.32E-02	1.33E+23	stable			
Rb-87	2.56E-02	2.59E+23	1.57E+18	4.41E-19	1.14E+05	1.14E+05
Rb-88	3.57E-02	3.61E+23	1.07E+03	6.48E-04	1.14E+15	0
Rb-89	4.72E-02	4.76E+23	9.19E+02	7.54E-04	1.51E+15	0
Rb-90	4.50E-02	4.54E+23	1.58E+02	4.39E-03	1.44E+15	0
Rb-90m	1.24E-02	1.25E+23	2.58E+02	2.69E-03	3.97E+14	0
Rb-91	5.58E-02	5.63E+23	5.80E+01	1.19E-02	1.78E+15	0
Rb-92	4.82E-02	4.86E+23	5.00E+00	1.39E-01	1.54E+15	0
Rb-93	3.55E-02	3.59E+23	6.00E+00	1.16E-01	1.14E+15	0
Rb-94	1.65E-02	1.67E+23	3.00E+00	2.31E-01	5.28E+14	0
Sr-87	2.56E-02	2.59E+23	stable			
Sr-88	3.57E-02	3.61E+23	stable			
Sr-89	4.73E-02	4.78E+23	4.37E+06	1.59E-07	1.51E+15	1.00E+15
Sr-90	5.78E-02	5.84E+23	9.09E+08	7.62E-10	3.95E+14	3.95E+14
Sr-91	5.83E-02	5.89E+23	3.47E+04	2.00E-05	1.86E+15	0
Sr-92	5.94E-02	6.00E+23	9.40E+03	7.37E-05	1.90E+15	0
Sr-93	6.24E-02	6.30E+23	4.46E+02	1.55E-03	2.00E+15	0
Sr-94	6.06E-02	6.12E+23	7.50E+01	9.24E-03	1.94E+15	0
Sr-95	5.27E-02	5.32E+23	2.40E+01	2.89E-02	1.69E+15	0
Sr-96	3.76E-02	3.79E+23	1.00E+00	6.93E-01	1.20E+15	0

Isotope	Cumulative Yield (/f)	Number of atoms	Half Life (s)	Decay Constant (/s)	Activity at end of life (Beq)	Activity at 30 days (Beq)
Sr-97	1.75E-02	1.77E+23	4.00E-01	1.73E+00	5.60E+14	0
Y-89	4.73E-02	4.78E+23	stable			
Y-90	5.78E-02	5.84E+23	2.30E+05	3.01E-06	1.85E+15	7.51E+11
Y-91m	3.38E-02	3.41E+23	2.98E+03	2.33E-04	1.08E+15	0
Y-91	5.83E-02	5.89E+23	5.06E+06	1.37E-07	1.86E+15	1.31E+15
Y-92	6.01E-02	6.07E+23	1.27E+04	5.46E-05	1.92E+15	0
Y-93m	2.21E-02	2.23E+23	8.00E-01	8.66E-01	7.08E+14	0
Y-93	6.35E-02	6.41E+23	3.66E+04	1.89E-05	2.03E+15	0
Y-94	6.45E-02	6.52E+23	1.12E+03	6.19E-04	2.06E+15	0
Y-95	6.38E-02	6.44E+23	6.18E+02	1.12E-03	2.04E+15	0
Y-96	6.00E-02	6.06E+23	5.00E+00	1.39E-01	1.92E+15	0
Y-96m	2.02E-02	2.04E+23	1.00E+01	6.93E-02	6.46E+14	0
Y-97	4.89E-02	4.94E+23	4.00E+00	1.73E-01	1.56E+15	0
Y-98	1.92E-02	1.94E+23	6.00E-01	1.16E+00	6.14E+14	0
Y-98m	1.11E-02	1.12E+23	2.00E+00	3.47E-01	3.54E+14	0
Y-99	2.08E-02	2.10E+23	1.50E+00	4.62E-01	6.66E+14	0
Zr-90	5.78E-02	5.84E+23	stable			
Zr-91	5.83E-02	5.89E+23	stable			
Zr-92	6.02E-02	6.08E+23	stable			
Zr-93	6.35E-02	6.41E+23	5.08E+13	1.36E-14	8.74E+09	8.74E+09
Zr-94	6.47E-02	6.54E+23	stable			
Zr-95	6.50E-02	6.57E+23	5.53E+06	1.25E-07	2.08E+15	1.50E+15
Zr-96	6.34E-02	6.40E+23	6.31E+26	1.10E-27	0.00E+00	0
Zr-97	5.98E-02	6.04E+23	6.03E+04	1.15E-05	1.91E+15	2.21E+02
Zr-98	5.64E-02	5.69E+23	3.10E+01	2.24E-02	1.80E+15	0
Zr-99	5.63E-02	5.69E+23	2.00E+00	3.47E-01	1.80E+15	0
Zr-100	5.58E-02	5.64E+23	7.00E+00	9.90E-02	1.79E+15	0
Zr-101	3.07E-02	3.10E+23	2.30E+00	3.01E-01	9.82E+14	0
Zr-102	2.05E-02	2.07E+23	3.00E+00	2.31E-01	6.56E+14	0
Nb-93	6.35E-02	6.41E+23	stable			
Nb-93m	6.03E-02	6.09E+23	5.09E+08	1.36E-09	6.73E+14	6.71E+14
Nb-95	6.50E-02	6.57E+23	3.00E+06	2.31E-07	2.08E+15	1.14E+15
Nb-97	6.00E-02	6.06E+23	4.33E+03	1.60E-04	1.92E+15	0
Nb-97m	5.63E-02	5.68E+23	5.90E+01	1.17E-02	1.80E+15	0
Nb-98	5.75E-02	5.81E+23	3.00E+00	2.31E-01	1.84E+15	0

Isotope	Cumulative Yield (/f)	Number of atoms	Half Life (s)	Decay Constant (/s)	Activity at end of life (Beq)	Activity at 30 days (Beq)
Nb-99	3.97E-02	4.01E+23	1.50E+01	4.62E-02	1.27E+15	0
Nb-99m	2.10E-02	2.12E+23	1.50E+02	4.62E-03	6.71E+14	0
Nb-100	3.11E-02	3.14E+23	1.50E+00	4.62E-01	9.95E+14	0
Nb-100m	3.11E-02	3.14E+23	3.00E+00	2.31E-01	9.95E+14	0
Nb-101	4.99E-02	5.04E+23	7.00E+00	9.90E-02	1.60E+15	0
Nb-102	2.84E-02	2.87E+23	4.30E+00	1.61E-01	9.08E+14	0
Nb-103	1.91E-02	1.93E+23	1.50E+00	4.62E-01	6.12E+14	0
Mo-95	6.50E-02	6.57E+23	stable			
Mo-97	6.00E-02	6.06E+23	stable			
Mo-98	5.79E-02	5.85E+23	stable			
Mo-99	6.11E-02	6.17E+23	2.37E+05	2.92E-06	1.95E+15	9.99E+11
Mo-100	6.29E-02	6.36E+23	2.30E+26	3.01E-27	0.00E+00	0
Mo-101	5.17E-02	5.22E+23	8.77E+02	7.90E-04	1.66E+15	0
Mo-102	4.28E-02	4.32E+23	6.78E+02	1.02E-03	1.37E+15	0
Mo-103	2.95E-02	2.98E+23	6.80E+01	1.02E-02	9.44E+14	0
Mo-104	1.79E-02	1.81E+23	6.00E+01	1.16E-02	5.72E+14	0
Tc-99	6.11E-02	6.17E+23	6.66E+12	1.04E-13	6.41E+10	6.41E+10
Tc-99m	5.38E-02	5.43E+23	2.16E+04	3.21E-05	1.72E+15	0
Tc-101	5.17E-02	5.22E+23	8.53E+02	8.12E-04	1.66E+15	0
Tc-102m	4.29E-02	4.33E+23	2.61E+02	2.66E-03	1.37E+15	0
Tc-103	3.03E-02	3.06E+23	5.40E+01	1.28E-02	9.70E+14	0
Tc-104	1.88E-02	1.90E+23	1.10E+03	6.30E-04	6.02E+14	0
Ru-99	6.11E-02	6.17E+23	stable			
Ru-101	5.17E-02	5.22E+23	stable			
Ru-102	4.30E-02	4.34E+23	stable			
Ru-103	3.03E-02	3.06E+23	3.39E+06	2.04E-07	9.70E+14	5.71E+14
Ru-104	1.88E-02	1.90E+23	stable			
Rh-103	3.03E-02	3.06E+23	stable			
Rh-103m	3.00E-02	3.03E+23	3.37E+03	2.06E-04	9.60E+14	0
Sb-131	2.56E-02	2.58E+23	1.38E+03	5.02E-04	8.18E+14	0
Sb-132	1.60E-02	1.62E+23	1.67E+02	4.15E-03	5.13E+14	0
Sb-132m	1.16E-02	1.17E+23	2.46E+02	2.82E-03	3.72E+14	0
Sb-133	2.40E-02	2.42E+23	1.40E+02	4.95E-03	7.67E+14	0
Te130	1.81E-02	1.83E+23	2.49E+28	2.78E-29	0.00E+00	0
Te131	2.55E-02	2.57E+23	1.50E+03	4.62E-04	8.16E+14	0

Isotope	Cumulative Yield (/f)	Number of atoms	Half Life (s)	Decay Constant (/s)	Activity at end of life (Beq)	Activity at 30 days (Beq)
Te132	4.29E-02	4.34E+23	2.77E+05	2.50E-06	1.37E+15	2.10E+12
Te133	3.06E-02	3.09E+23	7.50E+02	9.24E-04	9.78E+14	0
Te133m	3.99E-02	4.03E+23	3.32E+03	2.09E-04	1.28E+15	0
Te134	6.97E-02	7.04E+23	2.51E+03	2.76E-04	2.23E+15	0
Te135	3.34E-02	3.37E+23	1.90E+01	3.65E-02	1.07E+15	0
Te136	1.34E-02	1.35E+23	1.76E+01	3.94E-02	4.28E+14	0
I-131	2.89E-02	2.92E+23	6.93E+05	1.00E-06	9.25E+14	6.93E+13
I-132	4.31E-02	4.35E+23	8.26E+03	8.39E-05	1.38E+15	0
I-133	6.70E-02	6.76E+23	7.50E+04	9.24E-06	2.14E+15	8.50E+04
I-134	7.83E-02	7.91E+23	3.15E+03	2.20E-04	2.51E+15	0
I-135	6.28E-02	6.34E+23	2.37E+04	2.92E-05	2.01E+15	0
I-136	2.64E-02	2.67E+23	8.34E+01	8.31E-03	8.46E+14	0
I-136m	1.26E-02	1.28E+23	4.66E+01	1.49E-02	4.04E+14	0
I-137	3.07E-02	3.10E+23	2.45E+01	2.83E-02	9.82E+14	0
I-138	1.49E-02	1.50E+23	6.30E+00	1.10E-01	4.76E+14	0
Xe-131	2.89E-02	2.92E+23	stable			
Xe-132	4.31E-02	4.36E+23	stable			
Xe-133	6.70E-02	6.77E+23	4.53E+05	1.53E-06	2.14E+15	4.07E+13
Xe-134	7.87E-02	7.95E+23	1.83E+30	3.79E-31	0.00E+00	0
Xe-135	6.54E-02	6.60E+23	3.29E+04	2.11E-05	2.09E+15	0
Xe-135m	1.10E-02	1.11E+23	9.17E+02	7.56E-04	3.52E+14	0
Xe-136	6.31E-02	6.38E+23	6.83E+28	1.01E-29	0.00E+00	0
Xe-137	6.13E-02	6.19E+23	2.29E+02	3.03E-03	1.96E+15	0
Xe-138	6.30E-02	6.36E+23	8.48E+02	8.17E-04	2.01E+15	0
Xe-139	5.04E-02	5.09E+23	4.00E+01	1.73E-02	1.61E+15	0
Xe-140	3.65E-02	3.69E+23	1.36E+01	5.10E-02	1.17E+15	0
Xe-141	1.25E-02	1.26E+23	1.70E+00	4.08E-01	4.01E+14	0
Cs-133	6.70E-02	6.77E+23	stable			
Cs-135	6.54E-02	6.60E+23	7.26E+13	9.55E-15	6.30E+09	6.30E+09
Cs-137	6.19E-02	6.25E+23	9.49E+08	7.30E-10	4.07E+14	4.07E+14
Cs-138	6.71E-02	6.77E+23	1.95E+03	3.55E-04	2.15E+15	0
Cs-139	6.34E-02	6.41E+23	5.56E+02	1.25E-03	2.03E+15	0
Cs-140	5.72E-02	5.78E+23	6.37E+01	1.09E-02	1.83E+15	0
Cs-141	4.17E-02	4.21E+23	2.48E+01	2.79E-02	1.33E+15	0
Cs-142	2.72E-02	2.74E+23	1.70E+00	4.08E-01	8.70E+14	0

Isotope	Cumulative Yield (/f)	Number of atoms	Half Life (s)	Decay Constant (/s)	Activity at end of life (Beq)	Activity at 30 days (Beq)
Cs-143	1.45E-02	1.47E+23	1.80E+00	3.85E-01	4.65E+14	0
Ba-135	6.54E-02	6.60E+23	stable			
Ba-137	6.19E-02	6.25E+23	stable			
Ba-137m	5.85E-02	5.91E+23	1.53E+02	4.53E-03	1.87E+15	0
Ba-138	6.77E-02	6.84E+23	stable			
Ba-139	6.41E-02	6.48E+23	4.98E+03	1.39E-04	2.05E+15	0
Ba-140	6.21E-02	6.28E+23	1.10E+06	6.30E-07	1.99E+15	3.88E+14
Ba-141	5.83E-02	5.89E+23	1.10E+03	6.30E-04	1.87E+15	0
Ba-142	5.75E-02	5.81E+23	6.36E+02	1.09E-03	1.84E+15	0
Ba-143	5.54E-02	5.60E+23	1.45E+01	4.78E-02	1.77E+15	0
Ba-144	4.40E-02	4.44E+23	1.15E+01	6.03E-02	1.41E+15	0
Ba-145	1.93E-02	1.95E+23	4.30E+00	1.61E-01	6.18E+14	0
La-139	6.41E-02	6.48E+23	stable			
La-140	6.22E-02	6.28E+23	1.45E+05	4.78E-06	1.99E+15	8.30E+09
La-141	5.85E-02	5.91E+23	1.41E+04	4.91E-05	1.87E+15	0
La-142	5.85E-02	5.91E+23	5.47E+03	1.27E-04	1.87E+15	0
La-143	5.92E-02	5.98E+23	8.52E+02	8.13E-04	1.90E+15	0
La-144	5.47E-02	5.52E+23	4.08E+01	1.70E-02	1.75E+15	0
La-145	3.85E-02	3.89E+23	2.48E+01	2.79E-02	1.23E+15	0
La-146	1.67E-02	1.68E+23	6.10E+00	1.14E-01	5.33E+14	0
Ce-140	6.22E-02	6.28E+23	stable			
Ce-141	5.85E-02	5.91E+23	2.81E+06	2.47E-07	1.87E+15	9.87E+14
Ce-142	5.85E-02	5.91E+23	1.58E+24	4.39E-25	2.08E-01	0
Ce-143	5.96E-02	6.02E+23	1.19E+05	5.82E-06	1.91E+15	5.30E+08
Ce-144	5.50E-02	5.55E+23	2.46E+07	2.82E-08	1.76E+15	1.64E+15
Ce-145	3.93E-02	3.97E+23	1.81E+02	3.83E-03	1.26E+15	0
Ce-146	2.99E-02	3.02E+23	8.09E+02	8.57E-04	9.58E+14	0
Ce-147	1.89E-02	1.91E+23	5.64E+01	1.23E-02	6.04E+14	0
Ce-148	1.59E-02	1.61E+23	5.68E+01	1.22E-02	5.10E+14	0
Pr-141	5.85E-02	5.91E+23	stable			
Pr-143	5.96E-02	6.02E+23	1.17E+06	5.92E-07	1.91E+15	4.11E+14
Pr-144	5.50E-02	5.55E+23	1.04E+03	6.66E-04	1.76E+15	0
Pr-145	3.93E-02	3.97E+23	2.15E+04	3.22E-05	1.26E+15	0
Pr-146	3.00E-02	3.03E+23	1.45E+03	4.78E-04	9.59E+14	0
Pr-147	2.25E-02	2.27E+23	8.04E+02	8.62E-04	7.19E+14	0

Isotope	Cumulative Yield (/f)	Number of atoms	Half Life (s)	Decay Constant (/s)	Activity at end of life (Beq)	Activity at 30 days (Beq)
Pr-148	1.63E-02	1.65E+23	1.37E+02	5.06E-03	5.23E+14	0
Pr-149	1.07E-02	1.09E+23	1.36E+02	5.10E-03	3.44E+14	0
Nd-143	5.96E-02	6.02E+23	stable			
Nd-144	5.50E-02	5.55E+23	7.23E+22	9.59E-24	5.28E+00	5.28E+00
Nd-145	3.93E-02	3.97E+23	stable			
Nd-146	3.00E-02	3.03E+23	stable			
Nd-147	2.25E-02	2.27E+23	9.49E+05	7.30E-07	7.19E+14	1.08E+14
Nd-148	1.67E-02	1.69E+23	stable			
Nd-149	1.08E-02	1.09E+23	6.22E+03	1.11E-04	3.46E+14	0
Pm-147	2.25E-02	2.27E+23	8.28E+07	8.37E-09	6.68E+14	6.53E+14
Pm-149	1.08E-02	1.09E+23	1.91E+05	3.63E-06	3.46E+14	2.85E+10
Sm-149	1.08E-02	1.09E+23	stable			

Appendix C.2

This Appendix provides the gamma dose rate calculations for the fission products present at 30 days after reactor shutdown, as well as the 0.25% liquid remnant assumed to remain in the reactor vessel after draining the fuel/salt.

Isotope	Activity at 30 days (Beq)	Gamma Exposure rate constant (R m2/hr Ci)	Gamma dose at 1 meter from all FP (R/hr)	Gamma dose at 1 meter from 0.25% liquid remnant
Rb-87	1.14E+05	0.000	0.0	0.0
Sr-89	1.00E+15	0.000	0.0	0.0
Sr-90	3.95E+14	0.000	0.0	0.0
Y-90	7.51E+11	0.352	7.1	0.0
Y-91	1.31E+15	0.002	67.1	0.2
Zr-93	8.74E+09	0.000	0.0	0.0
Zr-95	1.50E+15	0.420	17049.4	42.6
Zr-97	2.21E+02	0.095	0.0	0.0
Nb-93m	6.71E+14	0.000	0.0	0.0
Nb-95	1.14E+15	0.434	13406.3	33.5
Mo-99	9.99E+11	0.083	2.2	0.0
Tc-99	6.41E+10	0.000	0.0	0.0
Ru-103	5.71E+14	0.282	4350.1	10.9
Te132	2.10E+12	0.115	6.5	0.0
I-131	6.93E+13	0.216	403.9	1.0
I-133	8.50E+04	0.343	0.0	0.0
Xe-133	4.07E+13	0.014	14.9	0.0
Cs-135	6.30E+09	0.892	0.2	0.0
Cs-137	4.07E+14	0.322	3542.6	8.9
Ba-140	3.88E+14	0.101	1060.4	2.7
La-140	8.30E+09	1.170	0.3	0.0
Ce-141	9.87E+14	0.000	0.0	0.0
Ce-143	5.30E+08	0.138	0.0	0.0
Ce-144	1.64E+15	0.000	0.0	0.0
Pr-143	4.11E+14	0.000	0.0	0.0
Nd-147	1.08E+14	0.000	0.0	0.0
Pm-147	6.53E+14	0.000	0.0	0.0
Pm-149	2.85E+10	0.005	0.0	0.0
Totals			39911	100