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A Feasibility Study for a Self-Sustained Reactor Fueled with Cintichem-type ^{99}Mo Targets

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A Feasibility Study for a Self-Sustained Reactor Fueled with Cintichem-type ^{99}Mo Targets

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Abstract

This report presents a comprehensive feasibility study for a unique and modified conventional reactor method to produce ^{99}Mo , the precursor of the most widely used medical radioisotope, $^{99\text{m}}\text{Tc}$. Utilization of a typical nuclear reactor requires the irradiation of stainless steel tubes called targets, coated with highly enriched uranium, to produce ^{99}Mo , a fission product of ^{235}U . These targets are typically surrounded by a driver core of fuel elements. The primary objective was to determine if a critical reactor could be constructed solely of Cintichem-type targets without the aid of a driver core. Additionally, this reactor would be capable of producing the entire U.S. demand for ^{99}Mo without compromising the Cintichem process, a chemical separation method used to extract ^{99}Mo from the irradiated uranium.

Numerous critical configurations were discovered, and additional reactor neutronic studies are presented which assess reactivity losses. Furthermore, an economic study was completed to estimate initial construction costs; and to provide a reliable comparison among the conventional, target-fueled, and solution-fueled reactor concepts in terms of waste disposal costs. Based on the neutronic and economic analyses, this new reactor concept appears quite feasible. Two reactor systems were chosen for further consideration as candidates in actual construction and experimental testing.

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EXECUTIVE SUMMARY

Over the past several years, numerous methods for the production of medical radioisotopes have been proposed, particularly those relating to the production of ^{99}Mo . The objective of this research is to evaluate the feasibility of a unique and alternative method for the production of ^{99}Mo . There are three major methods currently in use or under consideration:

- the conventional reactor method which utilizes the irradiation of targets;
- the solution reactor method which relies on a continuous batch process with on-line extraction capabilities; and
- target bombardment with the use of accelerators.

At Sandia National Laboratories, the Annular Core Research Reactor (ACRR) has been tasked by the U.S. Department of Energy to serve as the domestic source for ^{99}Mo production. This involves irradiating highly enriched uranium targets in the central region of the reactor using a driver core of fuel elements, and subsequently extracting the ^{99}Mo fission product from the uranium oxide coating inside the targets via the Cintichem process. The proposed technique utilizes the ACRR but does not require the use of a driver core. Instead, the reactor core consists solely of targets which serve the dual purpose of a fuel element and ^{99}Mo production source.

This research is to determine if a critical reactor core could be constructed with a reasonable number of targets that would still be able to meet the U.S. demand for ^{99}Mo . Los Alamos National Laboratory's Monte Carlo N-Particle transport code called MCNP (Version 4A) was utilized as the major analysis tool to model and evaluate the neutronics of this reactor. The project consisted of a neutronics and economic analysis composed of several major components:

1. construction of optimized combinations of moderator materials and target numbers which yield a critical system, and a parametric study on the variables which impact the system criticality such as target pitch, cladding materials, ^{235}U mass loading, and the number of targets in the core;
2. an analysis of temperature effects on k_{eff} and associated reactivity losses due to changes in moderator temperature and density as well as fuel temperature;
3. investigation of the effects of fission product poisoning from ^{135}Xe and ^{149}Sm , neutron poisoning from ^6Li production in beryllium-moderated systems, and associated reactivity losses;
4. fuel burnup analysis based on operation time and core size; and
5. an analysis of the economic viability of the proposed design with a comparison to other current and proposed methods in terms of construction and waste disposal costs.

The results of this investigation prove there are significant advantages associated with this reactor concept. The results demonstrate that the proposed system is a unique design that is both a neutronic and an economically viable option to ^{99}Mo production. This further validates that the reactor can meet or exceed the current ^{99}Mo demand for the United States while preserving the Cintichem process, thereby eliminating the need for modification of the current target design and extraction process. In addition, this concept is an improvement of the current conventional reactor method as the need to use a driver core, the large amount of spent fuel waste generated, and the associated costs of spent fuel storage and disposal are eliminated.

1. ISOTOPE PRODUCTION METHODS

1.1 Introduction

This research involves a comprehensive parametric analysis to determine the feasibility of ^{99}Mo production with a unique and alternative reactor method. This report will present the results obtained from parametric studies performed with the Monte Carlo N-Particle transport code MCNP (Version 4A) which was used to model a series of plausible optimum critical configurations with Cintichem-type targets.

An overview of the current state-of-the-art production methods for medical radioisotopes is given in Chapter 1. This includes an explanation of the nuclear formation mechanisms of ^{99}Mo ; current production methods such as conventional reactors, solution-fueled reactors, and accelerators; and a description of the proposed concept. Chapter 2 explains why a Monte Carlo code was chosen as the primary analysis tool, the parameters investigated, and the k_{eff} results obtained from the optimization of specific reactor configurations. Temperature and density effects on k_{eff} in terms of reactivity loss are summarized in Chapter 3 for a select number of critical configurations chosen from the optimized reactor configurations presented in Chapter 2.

Based on the critical configurations analyzed for temperature and density effects, three configurations were chosen from this list of candidates for further analysis. The first analysis evaluates the production of fission products and poisons. Primary fission products of concern here are ^{135}Xe and ^{149}Sm . For configurations containing beryllium as the moderator, effects from the production of ^6Li , helium gas, and tritium are also summarized. An analysis of the reactivity losses due to ^{235}U burnup for a given operation time is presented in Chapter 5. Finally, to quantify the neutronics results from an economic perspective, Chapter 6 describes an economic analysis including the approximate costs of construction and waste disposal for the proposed concept. These estimates are compared to estimates in the literature for current production methods. Appendices contain sample hand calculations, and a sample copy of an MCNP input and output file from a representative reactor configuration.

1.2 D.O.E. Isotope Production Project & History of ^{99}Mo Production

Medical isotope production was pioneered in the United States. The first $^{99}\text{Mo}/^{99\text{m}}\text{Tc}$ generator was developed by Brookhaven National Laboratory in the 1950s. Until 1966, the federal government was the main producer of ^{99}Mo using reactors at Brookhaven and Oak Ridge National Laboratories. At that time, the private sector developed the capability to produce isotopes on a commercial scale (Bragg, 1996); however, government reactors continued to serve as back-up production sources. Domestic private sector production was initially provided by the General Electric Test Reactor (GETR) in Pleasanton, California and the Cintichem, Inc. reactor in Tuxedo, New York. Production ceased at the GETR in 1977 when the reactor was decommissioned, leaving the United States with only one major production facility until 1989.

Production activities were suspended in 1989 at Cintichem when it was determined that costly reactor upgrades would have to be done to meet requirements by the Nuclear Regulatory Commission (NRC). Because of this and the reactor's age, the decommissioning option was subsequently chosen by Cintichem and began in December 1990. With the closure of the Cintichem reactor, the U.S. continues to rely on the National Research Universal (NRU) reactor in Chalk River, Ontario, Canada (operated by Nordion International) to provide approximately 60% of the supply of ^{99}Mo to pharmaceutical companies for distribution. The remaining 40% is supplied by a European source. Due to an impending vulnerability to the U.S. supply of ^{99}Mo and its reliance on foreign sources; Congress tasked the United States Department of Energy (DOE) to establish the Isotope Production and Distribution Program (IPDP). The mission of the IPDP is to establish a domestic ^{99}Mo production capability, ensure a reliable supply (up to 100% of the U.S. demand), and pursue eventual privatization of production (Carroll, 1997).

In November 1991, the DOE purchased the rights to the Cintichem technology to ensure continuity and stability in the production process, and to reduce the amount of time required for FDA approval. A comprehensive feasibility study was conducted to determine the most feasible DOE facility to serve as the production site. On completion of the study, the Annular Core Research Reactor and adjoining Hot Cell Facility (HCF) at Sandia National Laboratories were chosen as the reactor and processing site; and the Chemistry and Metallurgy Research Complex at Los Alamos National Laboratory was chosen to manufacture the irradiation targets. The Record of Decision (ROD) was issued by the DOE in October 1996, and physical modifications to the ACRR were completed a year later. However, physical modifications to the HCF are currently in progress and production activities are not slated to begin until summer 1999 at the earliest.

The NRU reactor has continued to supply the U.S. market for ^{99}Mo despite two labor disputes in June 1997 and May 1998 which temporarily shutdown the reactor and disrupted the supply. These events have created heightened awareness of the vulnerable state of the supply. Atomic Energy of Canada, Ltd. (AECL) of Canada has proposed and begun construction of two Maple reactors to take over the production from the NRU reactor when it is shutdown and decommissioned in the year 2000. However, this does not mitigate the possibility of future disruptions to the supply as these new reactors still require the U.S. to rely on a foreign source. It is hoped that the IPDP's efforts to establish a domestic production capability will be fruitful in providing an immediate production source with the ACRR and encourage the reestablishment of a private sector production facility.

1.3 Nuclear Formation & Medical Uses of ^{99}Mo / $^{99\text{m}}\text{Tc}$

^{99}Mo is the precursor to its daughter isotope, $^{99\text{m}}\text{Tc}$ which is the most widely used isotope for nuclear medical procedures in the United States today. About 36,000 nuclear diagnostic procedures are performed daily in addition to approximately 50,000 medicinal therapies as well as 100,000,000 lab tests per year with $^{99\text{m}}\text{Tc}$. This accounts for between 80 and 90% of all nuclear medicine procedures done annually in the U.S. (Parma, 1995).

The major reasons for its widespread use are due to its many valence states, low energy gamma (143 keV), and short half-life (6.01 hr). This reduces the amount of time the isotope resides in the body and minimizes the radiation dose to other parts of the body unaffected by the disease or ailment the isotope is being used to image. Medical imaging studies with ^{99m}Tc are done on the brain, bone, liver, spleen, and kidneys, as well as in blood flow studies. ^{99}Mo is shipped daily to pharmaceutical houses where ^{99m}Tc generators are produced. ^{99m}Tc is "milked" from the generator as the ^{99}Mo decays and is subsequently injected into the patient in various chemical forms. Compared to the time scale of long-lived isotopes, the half-lives of ^{99}Mo and ^{99m}Tc are short, about 66.0 hours and 6.01 hours, respectively. Thus it is imperative to continually produce the ^{99}Mo precursor to offset the loss of product due to radioactive decay, most of which occurs during processing and transportation. The current demand for the United States is approximately 19,700 Ci/wk at the source (including losses based a 30-hour processing and shipping time), or about 3,000 6-day Ci/wk at the pharmaceutical house dock (Parma, 1997). A 6-day Ci is defined as the amount of radioactive material received at the pharmaceutical house dock with a calculated decay time of six days. The source refers to the location where ^{99}Mo is produced. The world demand is about twice the U.S. demand.

^{99}Mo can be produced in a nuclear reactor either through the fission of ^{235}U or by neutron activation of ^{98}Mo . Fission production of ^{99}Mo requires the interaction of a neutron with a ^{235}U nucleus to induce the formation of fission fragments. The lighter of the two fission fragments is typically on the order of 95 AMU and can be up to 99 AMU which would correspond to a ^{99}Mo nucleus. In contrast, neutron activation requires a neutron to interact with a ^{98}Mo nucleus subsequently inducing an excitation reaction to form a ^{99}Mo nucleus. Figure 1 illustrates the nuclear formation of ^{99}Mo and the physical differences between fission production and neutron activation.

The formation of a ^{99}Mo nucleus occurs in approximately 6.1% of all possible fission events. Once a ^{99}Mo nucleus is formed via ^{235}U fission or neutron activation, two forms of ^{99}Tc are produced via beta emission. The most probable of these is the metastable isotope ^{99m}Tc , and its emission of a low energy gamma which are of primary interest to pharmaceutical medicine. After decaying with a short half-life of about 6 hours, the metastable state of ^{99m}Tc decays to a longer-lived form of ^{99}Tc and then ultimately to the stable isotope of ^{99}Ru . Figure 2 is an illustration of the ^{99}Mo decay chain and its daughter products.

Fission: $^{235}\text{U} (n,f) ^{99}\text{Mo}$
 $19,700 \text{ Ci } ^{99}\text{Mo} = 2.5 \times 10^{20} \text{ atoms } ^{99}\text{Mo} = 0.041 \text{ g}$

Activation: $^{98}\text{Mo} (n,\gamma) ^{99}\text{Mo}$
 $^{98}\text{Mo} = 24.13\%$ a natural abundance

Figure 1. Nuclear Formation of ^{99}Mo .

Figure 2. ⁹⁹Mo Decay Chain and Daughter Products.

There are several major reasons why the fission production of ⁹⁹Mo is preferred over neutron activation. The major disadvantages of neutron activation are:

- the low specific activity of ⁹⁹Mo generated;
- the requirement of large ^{99m}Tc generators;
- the generation of large amounts of spent fuel; and
- the need to use enriched ⁹⁸Mo.

In the ACRR for example, irradiation at a neutron flux level of 2×10^{13} n/cm²-sec generates a specific activity for fission produced ⁹⁹Mo of 75,000 Ci/g, whereas the specific activity generated through neutron activation is only 0.39 Ci/g, a difference in specific activity of about 200,000 times (Parma, 1995). This illustrates why a high power reactor with a large irradiation space would be needed to obtain a specific activity comparable to that of fission produced ⁹⁹Mo. Furthermore, such a large, high-power reactor would consequently produce large amounts of spent fuel. For a production level of 16,500 Ci/wk, approximately 47.7 kg of fully enriched ⁹⁸Mo would need to be irradiated (Parma, 1995). In comparison, 37 targets (each with a mass loading of 20 grams of ²³⁵U) could be irradiated at a low reactor power level to meet 113% of the total U.S. demand (Parma, 1997).

However, there are also disadvantages with using the fission process to produce ⁹⁹Mo. First, highly enriched uranium (HEU) on the order of 93% ²³⁵U must be used in the targets. This generates large amounts of low-level waste (LLW) which need to be reprocessed or disposed, in comparison to neutron activation which does not use uranium targets. Second, chemical processing must be used to separate the ⁹⁹Mo from the uranium whereas neutron activation does not require a chemical separation process. Unfortunately, a chemical extraction process can generate relatively large amounts of waste. However, unlike other extraction processes, the Cintichem process does not generate any mixed waste.

There are also trade-offs with the fission process which enhance the production of ^{99}Mo , but simultaneously complicate the waste disposition process. First, the power level in the target can be increased by an increase in the ^{235}U mass loading. For example, by adding 5 grams of ^{235}U to a 20 gram target, the power level in the target is increased by a factor of 1.25. Consequently, this increases the amount of LLW that must be disposed of, with a corresponding increase in the target coating thickness which is limited by the electroplating process and adherence issues (Parma, 1995). Second, the stainless steel cladding of the target decreases the fission production by up to 30%. To reduce this figure, a redesign of the current target would have to be done including changing the cladding to a less-absorbing material such as zirconium.

Research is currently being done at Argonne National Laboratory in Illinois to develop targets which utilize low enriched uranium (LEU) in the form of a thin foil (~125 μm thick) sandwiched between two slightly tapered inner and outer tubes of aluminum and zirconium, respectively (Hofman et al., 1997). To date, only thermal tests have been done with this concept, and it is unclear whether this would be a feasible alternative to the current HEU targets. Proliferation concerns have been expressed concerning the use of HEU to produce ^{99}Mo . However, there are some drawbacks to the use of LEU targets. First, it has not been determined whether the U.S. demand for ^{99}Mo can be met with the use of LEU targets. Second, the United States Food and Drug Administration (FDA) has not given approval to the method of ^{99}Mo production with LEU targets. Because there is more production of ^{239}Pu through (n,γ) reactions with the large amounts of ^{238}U in an LEU target, this presents a major problem in obtaining radionuclidic purity of the final product as required by the FDA and the pharmaceutical companies.

1.4 Target Irradiation & Extraction Processes

Several criteria must be met for ^{99}Mo to be produced by the fission process. First, the production facility must be able to handle and store relatively large quantities of Special Nuclear Material (SNM) in the form of targets. Second, the facility must have a reactor dedicated to production activities. Third, a hot cell must be present to process the targets, prepare the final product for shipment, and temporarily store solidified liquid waste. Fourth, a waste disposal site must be chosen to store radioactive waste generated. And finally, a distribution site must be nearby to provide timely access to transportation (Coats and Parma, 1994).

Target irradiation is typically the process used to produce ^{99}Mo in nuclear reactors. The Cintichem-type target which will be used in the ACRR consists of a stainless steel tube 18 inches (45.72 cm) long and 1.25 inches (3.18 cm) in diameter. The thickness of the tube is about 30 mils (0.0762 cm), and is coated by electroplating the inside with a thin layer of approximately 20 grams of ^{235}U in the form of 93% enriched UO_2 (Parma, 1998a). The top and bottom of the tube is sealed at both ends with welded end fittings, and the inside is back-filled with helium gas. The top end fitting allows the target to be handled remotely when removing it from the reactor, and the bottom end fitting has a hollow cylinder inside that acts as a catcher for any coating that flakes off from the inside of the tube. Figure 3 is a diagram of a typical Cintichem-type target with the appropriate dimensions. Each target is irradiated in the reactor core for 7 days (24 hrs/day). After the irradiation, several targets are removed from the reactor and are transferred to a hot cell for processing and preparation for shipping.

Figure 3. Schematic of a Typical Cintichem-type Target

Target irradiation in general requires a chemical extraction process to dissolve the uranium from the target and to precipitate the ^{99}Mo fission product from the uranium. There are several methods for chemically extracting ^{99}Mo produced from the fission process. The first is called the Cintichem process which was previously used by Cintichem, Inc. to produce ^{99}Mo and will be used by Sandia in its production activities. It is a complex but robust technique which relies on an acid cocktail to dissolve the uranium coating from the inside of a target and subsequently uses a precipitation process to extract the ^{99}Mo . The waste fission product-uranium solution, target, glassware, and associated equipment become radioactive waste which must be disposed.

Overall, the Cintichem process is very efficient and has some areas which can be further improved upon, but it does not generate the large amounts of waste associated with the UAI (uranium aluminide-aluminum dispersion) method used by Nordion to produce ^{99}Mo from irradiated targets (Parma, 1995). The second method for fission-produced ^{99}Mo is a fission fragment deposition method. This entails direct fission fragment deposition into the reactor moderator liquid or coolant gas that is eventually processed. A third method is similar to the second as it involves direct fission fragment deposition, except this process relies on the deposition to occur in the fuel which is at a high temperature and over time sublimates the leaking fission products into the coolant gas which is later processed.

1.5 Current Isotope Production Methods

The three major production methods described in the following sections are conventional reactors, solution-fueled reactors, and accelerator applications of ^{99}Mo production. General attributes of these methods as well as the advantages and disadvantages associated with each are discussed. These methods are compared to the proposed concept described at the end of this chapter.

1.5.1 Conventional Reactors

The use of a conventional nuclear reactor typically requires the irradiation of targets in a core of fuel elements which function as a driver core. The primary function of the driver core is to provide a secondary neutron source to the targets to aid in the production of ^{99}Mo via the fission process of ^{235}U . For example, the central region of the ACRR core will accommodate up to 37 targets for irradiation. Figure 4 illustrates the ACRR core prior to modification for ^{99}Mo production, and the modified core arrangement of targets and fuel elements.

The NRU reactor in Canada uses the target irradiation method to produce ^{99}Mo . There are other reactors throughout the United States which are also capable of producing ^{99}Mo . Among them are the ATR (Advanced Test Reactor) at the Idaho National Engineering & Environmental Laboratory in Idaho, the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory in Tennessee, and the Missouri University Reactor (MURR) at the University of Missouri-Columbia. These also use the target irradiation process, but instead either use modified targets which are slightly different than the typical Cintichem-type targets or irradiate fuel elements as in the case of MURR.

Current ACRR Configuration - 236 Fuel Elements

129 Fuel Element Baseline Core - 37 Targets

Figure 4. Cross Sectional View of the Original & Modified ACRR Core.

In general, target irradiation is the most commonly used and documented method for fission-produced ^{99}Mo today; however, the major problem with this method is the vast amounts of waste generated in production activities. The power level at which the reactor is operated, the number of fuel elements in the driver core, and the resulting fuel burnup determines the core lifetime and ultimately the amount of spent fuel generated. In addition, LLW is also generated from the processing of targets. Unfortunately, the large amounts of waste generated with conventional reactors pose a problem from an economic and storage standpoint. This is the main argument against the viability of the target irradiation method which has prompted the search for alternative methods.

1.5.2 Solution-Fueled Reactors

Solution-fueled reactors were among the first types of reactor systems investigated for power production purposes from the late 1940s through the 1960s. Among the first was the SUPO reactor at Los Alamos, a 25 kW reactor used to experimentally measure radiolytic decomposition of water. Recent proposals have been discussed and submitted regarding the use of this type of system to produce ^{99}Mo . To date, only small scale tests have been done with this type of reactor to determine its feasibility (Glenn et al., 1996); however, large scale tests have not been conducted to evaluate if it can handle full scale production. Advocates of this process claim the use of solution reactors can compete with or even outperform the target irradiation methods used in conventional reactors.

Babcock & Wilcox (B&W) Environmental Services, Inc. has proposed a passively cooled, solution reactor dedicated to ^{99}Mo production called the MIPR (Medical Isotope Production Reactor) (Ball, 1995). The reactor consists of a Zircaloy cylindrical vessel filled with a low-enriched ($\sim 19\%$ ^{235}U) aqueous solution of uranyl nitrate fuel, and is immersed in a pool of water which functions as a heat sink, reflector, and radiation shield. A power level of 200 kW has been proposed and estimated to produce approximately 2,000 Ci of ^{99}Mo daily. The reactor can be shutdown for about 1 hour daily to remove fuel for processing and extraction. The solution is passed through an ion exchange column to extract the ^{99}Mo , and the excess uranium and fission products are routed to a dump tank. This remaining excess solution is pumped back to the reactor vessel where it is reused for another production cycle.

Among the advantages cited by B&W for this reactor are its inherent safety characteristics (large negative power coefficient of reactivity), the ability to process and extract the ^{99}Mo online, continuous operation at a lower reactor power level than a conventional reactor, and a reduced inventory of fissionable material generation (Chopelas and Ball, 1993). The major advantage claimed by this concept is a significant reduction in the amount of waste generated because the fuel can be reused for additional production cycles as all of the fission product deposition occurs in the fuel. This eliminates the need to manufacture, purchase, and dispose of targets and fuel elements and their associated waste.

A second type of solution reactor proposed for ^{99}Mo production is based on a circulating solution reactor concept developed by Los Alamos National Laboratory called the KING reactor (Hills and Heger, 1997). The original intention of this reactor concept was to serve as an intense neutron source as it operates at a very high reactor power, approximately 25 MW. The reactor consists of an irradiation region filled with water in the central region of the core, the fuel region is contained in zirconium cladding outside the water region, and the outside of the core is surrounded by a beryllium reflector. The fuel consists of a 93.5 % ^{235}U enriched solution of uranyl sulfate, UO_2SO_4 with a fuel concentration varying between 86 to 220 g $^{235}\text{U/L}$.

It is estimated that if the reactor were operated at the nominal power of 25 MW that 15,000 Ci of ^{99}Mo could be produced in approximately 1.6 hrs (Hills and Heger, 1997). The major reason given for the capability to operate this type of a solution reactor at a much higher power as compared to the MIPR is the inherent heat capacity in a rapidly moving fuel solution. A second advantage cited is the elimination of an internal heat exchanger because the fuel can be cooled outside the reactor in a subcritical state.

In general, solution reactors seem to have an advantage over conventional reactors as they do not require the use of or generate waste with solid fuel assemblies and irradiation targets. However, there are several important issues which need to be investigated further with this production concept. Among these are heat removal, operating temperatures, and power level duration and limitations. There are currently no solution reactors operating on a regular basis in the U.S. in the tens of kW range. In comparison to the power levels proposed for the previously mentioned reactor concepts, the SUPO reactor's peak power was only 25 kW. A solution reactor in Taiwan has been identified as operating at a power of 100 kW but this has not been confirmed (Varley, 1994).

Other issues of concern are the recombination of O_2 and H_2 due to the radiolytic decomposition of water in the fuel, and the corrosive nature of the fuel solution on reactor components. Recombination systems are necessary with a solution reactor due to the large production amounts and the explosive nature of O_2 and H_2 . In pressurized systems, corrosion problems have been found to occur at $\sim 225^\circ\text{C}$ and phase stability problems at 300°C (Lane et al., 1958). Assuming successful solutions are found to the previously mentioned technical and safety problems, this type of production method still requires FDA approval which could prove to be lengthy and cumbersome.

1.5.3 Accelerators

In addition to conventional and solution fueled reactors, accelerator systems can also be used to produce ^{99}Mo . Accelerators primarily rely on spallation neutron reactions and charged particle interactions to achieve results equivalent to fission and neutron activation processes used in a reactor. There are numerous possible reaction combinations, but only a handful are feasible as their effectiveness is determined by cross section and threshold energy. Among these reaction mechanisms are neutron and proton activation, deuteron interaction, and fission reactions such as $^{98}\text{Mo}(n,\gamma)^{99}\text{Mo}$, $^{98}\text{Mo}(d,p)^{99}\text{Mo}$, $^{100}\text{Mo}(p,2n)^{99\text{m}}\text{Tc}$, $^{100}\text{Mo}(p,2p)^{99}\text{Mo}$, and $^{238}\text{U}(p,f)^{99}\text{Mo}$ (US DOE, 1995).

The Los Alamos Meson Physics Facility (LAMPF) accelerator at LANL has been used to conduct experimental ^{99}Mo production studies by bombarding natural uranium targets via a $^{238}\text{U}(\text{p},\text{f})$ reaction with high energy proton beams. Studies in the late 1980s showed that the accelerator was capable of producing 1,250 6-day Ci of ^{99}Mo per week by bombarding a natural uranium target with an 800 MeV proton beam at a current of 100 μA (Howe et al., 1996). Subsequent studies were conducted with lower energy beams of 200 and 400 MeV to determine if ^{99}Mo production is feasible with spallation and low-energy neutron interactions. Tests with a 200 MeV proton beam having a current of 1 μA incident on a target for 24 hours produced 62.06 Ci of ^{99}Mo (Waters and Wilson, 1996).

In addition, cyclotrons have also been used in studies with natural uranium targets. Cyclotrons accelerate protons to GeV energy levels. Research in this area has been done at the University of California-Davis (UCD) and test results have concluded that a ^{99}Mo yield of 17 Ci/hr is obtainable with proton-induced fission in ^{238}U targets (Lagunas-Solar et al., 1996a). The same group at UCD has proposed an alternative to the use of generators used with reactor-based methods by producing $^{99\text{m}}\text{Tc}$ directly with accelerators. This method uses a $(\text{p},2\text{n})$ reaction with enriched ^{100}Mo targets to produce the $^{99\text{m}}\text{Tc}$ directly without its precursor ^{99}Mo . Test results from this investigation have shown that a 1 hour bombardment of a target with a 1mA proton beam yields 23 Ci of $^{99\text{m}}\text{Tc}$ (Lagunas-Solar et al., 1996b).

However, there are two major disadvantages with the direct production method. First, direct production of $^{99\text{m}}\text{Tc}$ is not very practical due to its very short half-life of 6.01 hours, which presents transportation and production level limitations even with reactor methods. Second, the product was found to contain technetium-96 (4.35 days), a contaminant whose production level is directly proportional to proton energy. In theory, this method is possible; however, it could pose serious problems with obtaining FDA approval since the process falls short of meeting current U.S. production level demands and requires subsequent steps in the radiochemical processing to reduce product contamination.

Two other important drawbacks to accelerator produced ^{99}Mo are associated with the limitations of current accelerator technology. First, isotopes generated through nuclear transmutation generally have very low specific activities. Second, target heating becomes a serious problem with beam currents above several hundred μA . To meet 100% of the U.S. demand for ^{99}Mo , beam currents would have to be several orders of magnitude higher than those documented here (US DOE, 1995). Consequently, accelerator produced ^{99}Mo has been deemed for now to be neither technically feasible nor economical for meeting the current U.S. ^{99}Mo demand.

1.6 Proposed Concept

The three major methods presented previously have certain characteristics which make them feasible options for ^{99}Mo production. However, each has some disadvantages which limit their capability to produce ^{99}Mo on the commercial scale required by pharmaceutical needs. Consequently, other methods continue to be proposed. Conventional reactor methods are the most commonly used techniques for ^{99}Mo production, and this research focuses on eliminating the large volume of spent fuel and its associated costs for disposal. The main objective of this research was to determine if a unique alternative to the conventional reactor method could be devised which would be able to meet or exceed the U.S. demand for ^{99}Mo . The proposed concept is a modified version of the conventional reactor method with the major modification being use of the targets as the core replacing the usual driver core of fuel elements.

For the reactor concept to be a viable production option, several criteria had to be met:

- the core must consist of a reasonable number of targets to maintain criticality and production levels;
- reactor criticality must be maintained over its lifetime despite fuel burnup and poison production; and
- the system must be constructed and operated economically.

The proposed reactor core is composed of Cintichem-type targets arranged in a triangular pitch submerged in a moderator and coolant material. The exterior of the core is surrounded radially and axially with a reflector to reduce neutron leakage. Figure 5 illustrates a radial view of the reactor core of targets, and the reflector region. One of the major disadvantages of the conventional reactor is the large amount of waste generated by spent fuel elements. This translates into a significant volume of physical waste over the lifetime of a reactor, and a costly endeavor with regard to the manufacture, purchase, storage and ultimate disposal of the spent fuel elements. This proposed concept eliminates this waste volume because there is less uranium in the target than in a typical fuel element, and each target serves a dual function as a fuel element and a production source. The next chapter will detail the neutronics analysis and the results of the numerous parametric studies conducted to determine the optimized reactor system.

Figure 5. Radial View of a Reactor Fueled with Cintichem-type Targets.

2. NEUTRONICS ANALYSIS

This chapter summarizes the neutronics analysis, and explains the results obtained from the parametric studies conducted to determine optimized reactor configurations. The first section explains why the Monte Carlo approach was chosen over the deterministic codes to model and analyze numerous reactor configurations. Second, the moderator materials which were tested with various target configurations are described in terms of their neutronic and economic advantages. Third, the various parameters investigated with each reactor configuration are discussed. And finally, the construction and analysis methodologies of the configurations with their respective k_{eff} results are presented along with the major differences between each design.

2.1 Monte Carlo vs. Deterministic Codes

Numerous algorithms have been developed over the course of several decades to simulate particle transport in various media. The specific type of particle transport for this analysis is that of neutrons in a fissile system. The primary parameter of interest for measuring criticality in a fissile system is the effective multiplication factor (k_{eff}). k_{eff} is defined as the ratio of the neutron production gains relative to the losses incurred through absorption and leakage. To account for these processes, the Boltzmann transport equation is used to describe neutron transport. Consequently, to obtain a result for k_{eff} , this equation is solved as an eigenvalue problem.

Two major methodologies used in neutronics analyses to simulate neutron transport are the Monte Carlo and the deterministic methods. Several major differences are evident in how both methods determine the k_{eff} of a fissile system:

- the algorithm they use to obtain a result for k_{eff} ;
- differences in cross section data libraries;
- capabilities for modeling complex geometries; and
- computer time required to complete a given calculation.

Deterministic codes solve the Boltzmann transport equation directly to obtain an exact result for k_{eff} using the discrete ordinates method. This algorithm divides the space through which the particles travel into small boxes. k_{eff} is calculated by measuring the neutron balance in each differential portion of space. In contrast, Monte Carlo methods do not solve the transport equation directly; rather, they use a statistical approach to obtain a result for k_{eff} by simulating the stochastic behavior of individual particles and recording various aspects of this behavior over time. A Monte Carlo code utilizes a set of random numbers to generate a statistical history of each particle through what is called a random walk analysis. Random numbers are used at each interaction to determine what process the neutron will undergo: fission, absorption, or scattering. As each neutron is tracked throughout a media, the amount of energy lost or gained by the neutron, how many new neutrons are created from fission, and the direction of travel after an interaction are among the measured parameters.

Another difference is the format of neutron cross section data libraries utilized in both codes. A neutron cross section is the probability that a neutron will undergo a certain interaction in its lifetime such as fission, scattering, or absorption. The random numbers generated by the code are directly correlated to the cross sections in the data library and are used in the calculation to determine how a neutron will behave in various materials. Deterministic codes generally rely on multigroup cross sections whereas Monte Carlo codes typically utilize continuous energy cross section data. Some Monte Carlo codes are also capable of using multigroup data. Multigroup cross sections average the probability of interaction with respect to a flux weighted spectrum over a specific range of energy bins. Consequently, the flux spectrum of the modeled system must match the spectrum of the multigroup data set being used for the calculation to be correct. In comparison, continuous energy cross sections are not averaged according to a spectrum but are instead point-wise continuous over a wide range of energies and thus can be applied to a wider variety of problems.

A major advantage of Monte Carlo codes is their ability to model and analyze complex, heterogeneous geometries. For a deterministic code to analyze the same geometry, the geometry must be simplified or homogenized. However, a disadvantage of Monte Carlo codes is the amount of computer time required for a calculation. This is generally a function of the complexity and composition of the modeled geometry, the number of neutron generations simulated, and the number of neutrons per generation the code is required to track. Deterministic methods are generally quicker because they are limited to simplified geometries, utilize simpler cross section data sets, and solve a mathematical equation directly rather than simulating neutron behavior.

The main tool used in the neutronics analysis for this project was Los Alamos National Laboratory's Monte Carlo N-Particle transport code called MCNP (Version 4A) (Briesmeister, 1993). MCNP is a robust and versatile code that is widely used for neutronics analyses; in recent years, its use has expanded worldwide. There are several reasons why MCNP was chosen over deterministic codes to construct and simulate this reactor concept. First, utilization of continuous energy cross sections proved more advantageous than using multigroup data because the neutron spectra of the modeled configurations were not explicitly determined. Using this type of cross section data reduced the possibility of modeling a system with a different spectrum than that of the cross section library.

Second, modeling a heterogeneous system of targets and moderator with a deterministic code requires physical and calculational homogenization of the core. With the numerous parametric studies conducted, this would introduce additional error with respect to ensuring the core was properly homogenized and the corresponding atom densities were used. Finally, the required computer time for a Monte Carlo calculation has been significantly reduced with the advent of faster computers in recent years. It has become a more viable option considering the recent advances in computer speed and technology.

2.2 Moderator Materials

Reactors generally require the use of a special material called a moderator to thermalize or slow down fission neutrons. In some reactors, the moderator also serves a dual role as the coolant material to ensure proper and efficient removal of fission generated heat from the core. For a material to be considered a viable choice for a moderator, it must possess several characteristics: a low mass number, a small absorption cross section, and a large scattering cross section. However, no specific material possesses all of these traits so some trade-offs must be considered when choosing a moderator for the reactor design. For the proposed concept, different moderators were tested in the parametric analyses with MCNP: water; beryllium, beryllium oxide, graphite, and heavy water.

Most reactors are moderated and cooled with ordinary water. It is relatively inexpensive, has excellent slowing down properties, has a small migration length for thermal neutrons, and can also be used as the reactor coolant. However, for a system to attain and maintain criticality, enriched uranium fuel must be used. The use of enriched fuel provides for a small reactor size due to the small thermal neutron migration length. Beryllium has a low absorption cross section but a reasonable scattering cross section. A major neutronic advantage for maintaining criticality with beryllium is its ability to produce $(n,2n)$ reactions. However, its poor mechanical properties, low thermal conductivity at high temperatures, and high cost to fabricate make its use generally unfavorable. Beryllium metal is very brittle and this is enhanced at high temperatures due to the formation of helium gas in the voids of the metal through (n,α) reactions. Beryllium oxide has neutronic properties very similar to beryllium, is easier to fabricate, and generally less expensive. Unfortunately, it vaporizes rapidly in the presence of moisture at high temperatures and leads to the formation of beryllium hydroxide. A major drawback with beryllium and its compounds are their extreme toxicity to the body when inhaled.

Graphite is the most widely used of the solid moderators. It has good neutronic properties but not as good as those of beryllium. It is less expensive to fabricate than beryllium due to good mechanical and thermal properties. It also has good fission product gas retention properties which make it comparable to the containment characteristics of typical fuel element cladding. Unfortunately, graphite has a tendency to contract and expand at high irradiation temperatures due to high neutron fluence and the anisotropic nature of its atomic structure.

Heavy water has relatively good slowing-down properties, a very low absorption cross section, and a reasonable scattering cross section. This enhances the resonance escape probability and thermal utilization of the fuel. Another major advantage of heavy water is that natural uranium can be used as fuel, enriched fuel is not required for operation as it is in a water moderated reactor. However, heavy water is expensive, rather limited in distribution, and requires pressurization if it is used as a coolant in addition to a moderator.

2.3 Parametric Investigations

To create a feasible reactor configuration, it was necessary to perform parametric studies to determine optimum combinations of targets and moderator. The targets and reactor vessel were modeled with MCNP in three dimensions. Each target consisted of a series of concentric cylinders with planes at the top and bottom to give the target a finite height. A reactor core was constructed by encasing the targets in a hexagonal lattice to arrange them on a triangular pitch. Figure 6 is a two-dimensional schematic showing a radial and axial view of the target with the moderator surrounding the target's components as modeled in MCNP. The thicknesses of the UO_2 layer and stainless steel cladding in Figure 6 are thicker than those in an actual target and exaggerated here for clarity.

Figure 6. Radial and Axial Views of a Single Cintichem-type Target Modeled in MCNP.

MCNP Target Model

Actual Cintichem-type Target

Figure 7. Cross Sectional Views of MCNP Modeled and Nominal Cintichem-type Targets.

Figure 6 also illustrates an axial view of the target with the top and bottom end caps omitted from the model. These end caps were omitted assuming the k_{eff} results would not be compromised by a gross over-simplification of the geometry. Figure 7 is a cross sectional view illustrating the differences between the modeled target and an actual target. It was determined that adding the end caps to the targets would have created a more complex model. In addition, the resulting k_{eff} s would have been less conservative because the additional stainless steel creates more neutron absorption.

To gain a comprehensive understanding of the dynamics which affect this type of system, several major parameters were investigated in the analysis. Among these were:

- number of targets that would yield a critical system;
- optimum moderator material;
- ^{235}U target mass loading;
- target pitch or spacing; and
- a comparison of the nominal stainless steel cladding to a material with less neutron absorption such as zirconium.

The baseline number of targets was 37, which corresponds to the maximum number of targets that will fit into the central region of the ACRR. This number of targets was chosen for two reasons. First, this number of targets gives a basis from which to construct a reactor and compare to the ACRR. Second, assuming an irradiation schedule of seven days, this would correspond to a processing rate of between five or six targets per day to meet the U.S. demand.

For the five moderators tested, various reactor configurations were constructed via an incremental methodology through three parametric optimization studies. The first study consisted of constructing a reactor core with targets infinite in height surrounded by a infinite radial reflector of the moderator material being investigated. This study was done to determine the optimum target pitch. For the solid moderators, the target pitch and coolant channel radius were optimized simultaneously. The optimum coolant channel radius was obtained by adding 0.20 cm increments of water around each target beginning with a minimum of 0.0 cm up to a maximum of 1.0 cm.

The second study consisted of finding the optimum radial reflector thickness. This was done by keeping the targets infinite in height and reducing the radial reflector in incremental steps from an infinite size (two neutron diffusion lengths) to a bare system. For the solid moderators and heavy water, an infinite reflector of ordinary water was put on the outside to further reduce neutron leakage.

The third and final study consisted of optimizing the axial reflector thickness. This was done by using targets of finite height arranged on a defined pitch surrounded by a finite radial reflector determined by the previous two studies. The optimum axial reflector thickness was determined in the same incremental manner as the second study. An infinite reflector of water was added to the outside of the optimized reflector to reduce neutron leakage in the axial direction. The following section will summarize and explain the k_{eff} results obtained from the configurations constructed with the above methodology in MCNP. A summary of the optimized dimensions for each reactor configuration will be characterized for each moderator in terms of the k_{eff} results. In addition, reasons for modifications to the nominal target design such as the addition of solid plugs in the target central region will also be discussed. These studies pertain to conditions at 27 °C (300 K) and were evaluated with appropriate cross sections for this temperature.

2.4 Reactor Design Results

2.4.1 k_{eff} Verification Cases

MCNP (Version 4A) was run on several personal computers to expedite the process of executing several hundred input files comprising the numerous parametric studies done for this analysis. This version of the code was previously compiled with the Lahey Fortran Compiler, Version 4.1L to run on a personal computer. All of the machines used were stand alone mode, IBM compatible computers with Pentium processors of various speeds. Three input files were chosen to verify that the code is machine independent, and that no significant differences should be expected in the final k_{eff} and standard deviation results regardless of the machine used. A heterogeneous case was tested to determine if the geometric complexity contributed to any difference in the final k_{eff} depending on the machine used. Two homogeneous cases were chosen because they are simple models that yield a k_{eff} result quickly.

The first input file models a heterogeneous reactor core comprised of 85 Cintichem-type targets arranged in a triangular pitch arrangement in a water moderator with 10 cm radial and axial water reflectors. Each target was set on a pitch of 3.95 cm, contained a ^{235}U mass loading of 30.0 grams, a stainless steel cladding thickness of 30 mils (0.0762 cm), and the central void filled with a beryllium plug. The remaining cases modeled two fast, critical assemblies located at Los Alamos National Laboratory. Both of these reactor systems are considered homogenous systems as they are composed only of fuel without moderator. The first of these files modeled the Godiva critical assembly. It consists of a bare, 93.71% enriched ^{235}U sphere with a radius of 8.741 cm. The second fast system modeled was the Jezebel critical assembly. This is a bare, plutonium sphere consisting of a mixture of ^{239}Pu , ^{240}Pu , ^{241}Pu , and approximately 1% gallium. The sphere has a radius of 6.384 cm with a 0.013 cm thick outer nickel shell.

Table 1 presents the k_{eff} results and associated standard deviations for the three cases run on five different machines. Each case was run with 400,000 particle histories. No significant differences in k_{eff} and the standard deviations were obtained among the five machines tested for the Godiva and Jezebel assemblies. However, slight differences are evident when comparing machines 3 & 5 to machines 1, 2, and 4. These deviations may be attributable to differences in the hardware architectures of the computer systems. Consequently, this test proves there is consistency in the k_{eff} and associated standard deviation results obtained from these machines for the reactor optimization analysis.

Table 1. k_{eff} Results of MCNP 4A Verification Cases.

Machine Type	Case 1 85 SS targets, water core w / Be plugs, 30 g ^{235}U	Case 2 Godiva assembly (bare ^{235}U sphere)	Case 3 Jezebel assembly (bare Pu sphere)
Dell Optiplex GXMT 5133 (133 MHz)	1.02621 $\pm$ 0.00130	0.99398 $\pm$ 0.00092	1.00628 $\pm$ 0.00095
Compaq Presario 9546 (100 MHz)	1.02621 $\pm$ 0.00130	0.99398 $\pm$ 0.00092	1.00628 $\pm$ 0.00095
Access Technologies ATC LXP2 (266 MHz)	1.02611 $\pm$ 0.00133	0.99398 $\pm$ 0.00092	1.00628 $\pm$ 0.00095
Dell Optiplex GX1 (400 MHz)	1.02621 $\pm$ 0.00130	0.99398 $\pm$ 0.00092	1.00628 $\pm$ 0.00095
Gateway 2000 P5/90 (90 MHz)	1.02611 $\pm$ 0.00133	0.99398 $\pm$ 0.00092	1.00628 $\pm$ 0.00095

2.4.2 Water Moderator

The first set of designs were constructed with Cintichem-type targets in a reactor vessel filled with water moderator. An axial and radial reflector of water 10 cm in thickness completely surrounds each reactor core. Table 2 presents the k_{eff} and standard deviation results for the twenty water moderated configurations evaluated.

Table 2. MCNP Results for Water Moderated Reactors

Target Pitch (cm)	Total Number of Targets*	²³⁵ U Mass (grams)	Cladding Material	Target Center	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	k _{eff} ± σ
4.50	85	28.2	SS	Dry	H ₂ O / H ₂ O	10.0 / 10.0	0.79410 ± 0.00228
4.50	85	28.2	Zr	Dry	H ₂ O / H ₂ O	10.0 / 10.0	0.91593 ± 0.00274
4.50	163	28.2	SS	Dry	H ₂ O / H ₂ O	10.0 / 10.0	0.90227 ± 0.00235
4.50	163	28.2	Zr	Dry	H ₂ O / H ₂ O	10.0 / 10.0	1.03275 ± 0.00208
3.50	85	28.2	SS	H ₂ O	H ₂ O / H ₂ O	10.0 / 10.0	0.98528 ± 0.00217
3.50	85	28.2	Zr	H ₂ O	H ₂ O / H ₂ O	10.0 / 10.0	1.12514 ± 0.00221
3.95	37	28.2	SS	Be	H ₂ O / H ₂ O	10.0 / 10.0	0.80375 ± 0.00257
3.90	37	28.2	Zr	Be	H ₂ O / H ₂ O	10.0 / 10.0	0.91850 ± 0.00253
3.95	85	28.2	SS	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.00938 ± 0.00125
3.90	85	28.2	Zr	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.16319 ± 0.00141
3.00	85	28.2	SS	Poly	H ₂ O / H ₂ O	10.0 / 10.0	1.00947 ± 0.00226
3.50	85	28.2	Zr	Poly	H ₂ O / H ₂ O	10.0 / 10.0	1.14135 ± 0.00285
3.95	85	30.0	SS	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.02621 ± 0.00130
3.90	85	30.0	Zr	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.17403 ± 0.00145
3.95	85	20.0	SS	Be	H ₂ O / H ₂ O	10.0 / 10.0	0.91169 ± 0.00114
3.90	85	20.0	Zr	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.08903 ± 0.00139
3.95	61	30.0	SS	Be	H ₂ O / H ₂ O	10.0 / 10.0	0.94628 ± 0.00122
3.90	61	30.0	Zr	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.07758 ± 0.00139
3.95	61	20.0	SS	Be	H ₂ O / H ₂ O	10.0 / 10.0	0.83864 ± 0.00113
3.90	61	20.0	Zr	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.00056 ± 0.00142

*All targets used 93% enriched uranium.

k_{eff} results in bold face type denote critical reactor cores. The results illustrate that only a limited number of practical reactor combinations can be constructed in a water moderated system. This is primarily due to the strong neutron absorption characteristics of the stainless steel cladding. Most of the configurations composed of zirconium clad targets yielded a critical system.

The first set of designs consisted of targets with stainless steel or zirconium cladding 30 mils (0.0762 cm) in thickness, a mass loading of 28.2 grams of ²³⁵U, and a dry central region. A reactor containing zirconium clad targets was the only configuration with a pitch of 4.50 cm to attain criticality. Based on the results of these cases, construction of a reactor with nominal Cintichem-type targets, especially targets with stainless steel cladding was deemed to be highly impractical and insufficient as a viable design choice.

A second parametric study was conducted to determine alternatives which could be made to the nominal target design to attain criticality with a reasonable number of targets. Adding more targets and increasing the mass loading of ²³⁵U in the target was determined to be counterproductive as this would increase k_{eff} while simultaneously increasing the volume of waste generated later. Thus it was determined that adding a moderator material to the central region would be a more efficient alternative to offset the absorption by the stainless steel cladding.

A filled center region was simulated with three moderators: water, beryllium, and polyethylene. Water would be used to flood the target center. Beryllium and polyethylene would be cast in the form of solid plugs which would fit into the central region and be reused for subsequent irradiation cycles. Figure 8 compares the k_{∞} results from the target pitch optimization study for 85 stainless steel targets using these fill materials to a target with a dry target center. Figure 9 is a comparison of the k_{∞} results as a function of target pitch for 85 zirconium clad targets.

Figure 8. k_{∞} Results for Water Moderated Reactors Using 85 Stainless Steel Clad Targets and Various Target Center Materials.

Figure 9. k_{∞} Results for Water Moderated Reactors Using 85 Zirconium Clad Targets and Various Target Center Materials.

Higher k_{∞} values were obtained for zirconium clad than for the stainless steel targets, regardless of the central region contents. Additionally, the optimum pitch was smaller in cases with a filled center than with the nominal dry center target. Generally, optimum pitches for filled targets were 0.50 to 1.50 cm smaller than those for dry targets depending on the fill material.

A pair of reactors with 37 targets were evaluated using both cladding materials and beryllium plugs. However, these cases yielded very subcritical systems so designs with 85 targets were further evaluated. For stainless steel targets, the use of beryllium plugs yielded slightly higher k_{eff} values than those with polyethylene plugs. High k_{eff} values were obtained for these moderators because of their scattering properties and the (n,2n) reactions in beryllium. For zirconium clad targets, beryllium was also determined to be the best plug material. Polyethylene plugs and a water flooding were alternative moderators of choice for the central region in a zirconium clad target.

Additional reactor configurations were also analyzed using beryllium plugs in the target center for stainless steel and zirconium targets. The mass loadings in the targets for these cases were 20 and 30 grams of ^{235}U . For these target loadings, reactor cores consisting of 61 and 85 targets were investigated. Target cores consisting of 61 stainless steel targets with 20 and 30 grams of ^{235}U were below a k_{eff} of 0.95. In contrast, reactor cores with 61 zirconium targets for both ^{235}U mass loadings were above a k_{eff} of 1.00. All configurations evaluated with 85 targets containing 30 grams of ^{235}U were above a k_{eff} of 1.00. For the same number of targets loaded with 20 grams of ^{235}U , only the zirconium targets were above a k_{eff} of 1.00. The stainless steel targets were subcritical and near a k_{eff} of 0.90.

Unfortunately, there are several disadvantages with using a moderator in the target central region. Flooding the internal region of a target with water could cause problems with ^{99}Mo extraction and the Cintichem process. In addition, a flooded target would need to be pressurized to prevent a rupture from occurring. Beryllium is an excellent moderator but it is expensive and difficult to fabricate. Questions concerning the effects of radiation on beryllium could affect the plugs reuse after an irradiation cycle. Polyethylene is relatively inexpensive and easy to fabricate. However, polyethylene is prone to neutron degradation which could also limit its reuse.

Critical configurations in a water moderated system are very limited and require modifications to the nominal target design. A critical system with the nominal target design and water moderation cannot be attained with less than 163 targets. Utilizing the central region of the target as an internal moderator for fission neutrons leaving the UO_2 layer proved to be a successful concept in offsetting the strong neutron absorption by the stainless steel cladding. It also reduced the required number of targets for criticality by nearly half from 163 to 85. Consequently, a moderator material must be used in a target for a water moderated system to be a viable option. Furthermore, requiring the construction of a core larger than the baseline 37 targets is unrealistic because only a small fraction of the targets could be used for processing. A core of a large size would require leaving some targets in the reactor for an extended amount of time thereby creating a driver core, which essentially defeats the original purpose of this reactor concept.

2.4.3 Beryllium & Beryllium Oxide Moderators

The second set of moderator materials investigated were beryllium and beryllium oxide. Twelve configurations were investigated with beryllium; two with beryllium oxide. In addition, two configurations were evaluated to determine if the proposed concept would work with low enriched targets. These targets were evaluated with a ^{235}U enrichment of 20 weight percent. Parametric analyses were performed for each reactor configuration to determine the optimum target pitch, target coolant channel size, and dimensions of the axial and radial reflectors. The exterior of the reactor was surrounded by a 10 cm water reflector in the radial and axial directions to further reduce neutron leakage. Because beryllium and beryllium oxide are solid moderators, it was necessary to analyze the neutronic effects of a water target coolant channel and optimize its size concurrently with the target pitch.

Table 3. MCNP Results for Beryllium & Beryllium Oxide Moderated Reactors

Target Pitch (cm)	Total Number of Targets*	²³⁵ U Mass (grams)	Cladding Material	Target Center	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	$k_{eff} \pm \sigma$
7.00	91	28.2	SS	Dry	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.09316 ± 0.00256
7.00	91	23.5	SS	Dry	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.03380 ± 0.00278
7.00	37	28.2	SS	Dry	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	0.92340 ± 0.00233
7.00	37	23.5	SS	Dry	Be / H ₂ O	30.0 / 30.0 Be 10.0 / 10.0 H ₂ O	0.86986 ± 0.00226
7.00	37	28.2	Zr	Dry	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.10859 ± 0.00295
7.00	37	23.5	Zr	Dry	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.06742 ± 0.00288
6.00	37	28.2	SS	Poly	Be / H ₂ O	30.0 / 15.0 Be 10.0 / 10.0 H ₂ O	1.01514 ± 0.00308
5.00	37	28.2	Zr	Poly	Be / H ₂ O	30.0 / 15.0 Be 10.0 / 10.0 H ₂ O	1.18341 ± 0.00307
7.00	37	28.2	SS	ZrH(1.85)	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	0.96842 ± 0.00268
6.00	37	28.2	Zr	ZrH(1.85)	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.13868 ± 0.00324
6.00	37	28.2	SS	Be	Be / H ₂ O	25.0 / 20.0 Be 10.0 / 10.0 H ₂ O	1.00458 ± 0.00252
6.00	37	28.2	Zr	Be	Be / H ₂ O	25.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.19729 ± 0.00263
8.00	37	28.2	SS	Dry	BeO / H ₂ O	30.0 / 25.0 BeO 10.0 / 10.0 H ₂ O	0.91495 ± 0.00245
8.00	37	28.2	Zr	Dry	BeO / H ₂ O	30.0 / 35.0 BeO 10.0 / 10.0 H ₂ O	1.10428 ± 0.00296

*All targets used 93% enriched uranium and had a 0.20 cm gap for water cooling.

Table 3 presents the k_{eff} and standard deviation results for beryllium and beryllium oxide moderated reactors cooled with water. k_{eff} results in bold face type denote critical cores. The concept of using solid plugs in the central region was employed in this study to quantify its effects on k_{eff} in beryllium and beryllium oxide moderated systems. A base case was defined as a target with a dry center, and its k_{eff} was compared to those for three moderating materials tested as solid plugs: beryllium, polyethylene, and zirconium hydride (ZrH_{1.85} - 64.9 % hydrogen).

The first beryllium configurations were composed of 37 targets with dry central regions at two ²³⁵U mass loadings of 23.5 and 28.2 grams. The optimum pitch for stainless steel and zirconium targets was determined to be 7.00 cm with a coolant water gap size of 0.20 cm. k_{eff} results for 37 stainless steel targets using both mass loadings were below 0.95. In contrast, k_{eff} results obtained for reactor cores composed of 37 zirconium targets were above 1.05. As 37 stainless steel targets were subcritical, subsequent studies determined 91 stainless steel targets were required for a critical system at both ²³⁵U mass loadings. Consequently, a reactor core of stainless steel targets with dry central regions moderated by beryllium was deemed impractical for further design consideration.

Subsequent studies were done with beryllium, polyethylene, and zirconium hydride plugs. The number of targets modeled here was 37 with a mass loading of 28.2 grams of ^{235}U per target. Based on studies to determine the optimum pitch for targets with plugs, pitches for target centers with solid plugs were found to be 1.00 to 2.00 cm smaller than the 7.00 cm pitch for the dry center targets. However, the optimum water coolant channel thickness remained the same at 0.20 cm. Figure 10 illustrates k_{eff} results for the optimized reactors using stainless steel targets, moderated and reflected with beryllium. Figure 11 shows the k_{eff} results for the reactors using zirconium clad targets.

Figure 10. k_{eff} Results for Beryllium Moderated Reactors Using 37 Stainless Steel Clad Targets and Various Target Center Materials.

Figure 11. k_{eff} Results for Beryllium Moderated Reactors Using 37 Zirconium Clad Targets and Various Target Center Materials.

Polyethylene plugs yielded slightly higher k_{eff} values than beryllium plugs in stainless steel targets. Unfortunately, k_{eff} values using both of these types of plugs were barely above 1.00. ZrH_{1.85} plugs yielded a subcritical system with a k_{eff} of 0.96. All of the cases for zirconium targets were above a k_{eff} of 1.10, with beryllium plugs being the moderator of choice followed by polyethylene and ZrH_{1.85} plugs.

Two configurations were also designed with beryllium oxide as the moderator. These reactors used 37 targets (28.2 grams of ²³⁵U per target) with stainless steel and zirconium claddings. Both reactors were designed to use dry targets without solid plugs in the target central region. In comparison to the same cases using beryllium as the moderator, the k_{eff} s for these cases were slightly lower due to absorption by the oxygen. Further studies with beryllium oxide were suspended as these results confirmed its neutronic properties are similar to those of beryllium.

The results presented thus far show that highly enriched targets work very well in obtaining a critical system. However, proliferation concerns have been expressed in recent years with the use of highly enriched uranium to produce ^{99}Mo . Capabilities do not currently exist to produce ^{99}Mo with low enriched uranium on a commercial scale and are still in the development phase. For the purpose of this research, two configurations were designed to evaluate the feasibility of this reactor concept.

Both reactors modeled for the low enriched studies contained 37 targets and a mass loading of 28.2 grams of ^{235}U per target. Zirconium was chosen as the cladding material due to its low neutron absorption characteristics and the large number of critical cases produced with highly enriched targets. Beryllium was chosen as the moderator and plug material for its ability to produce $(n,2n)$ reactions and neutron scattering. Table 4 presents the results for the low enriched target reactors. k_{eff} results for both reactors are above 1.05. The addition of a plug in the central region of the target also provides a significant increase in k_{eff} . These results demonstrate that low enriched targets can attain a critical configuration in beryllium moderated systems with a 37 target core. However, future work would need to be done to assess this capability with other moderators.

A number of critical configurations were obtained in beryllium moderated systems. In comparison to water moderated reactors, there are significant possibilities with beryllium for constructing a system with only 37 targets. All of the cases investigated with zirconium targets were critical regardless of whether the central region of the target was dry or contained a solid plug. Reactor cores with 37 stainless steel targets are also possible; however, they are limited to using a solid plug. The 91 dry stainless steel targets required for criticality creates a much larger core needed to satisfy full scale production.

Table 4. MCNP Results for Low Enriched Beryllium Moderated Reactors

Target Pitch (cm)	Total Number of Targets*	^{235}U Mass (grams)	Cladding Material	Target Center	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	$k_{\text{eff}} \pm \sigma$
7.00	37	28.2	Zr	Dry	Be / H_2O	30.0 / 25.0 Be 10.0 / 10.0 H_2O	1.07600 ± 0.00325
7.00	37	28.2	Zr	Be	Be / H_2O	30.0 / 30.0 Be 10.0 / 10.0 H_2O	1.14517 ± 0.00269

*All targets used 20% enriched uranium and had a 0.20 cm gap for water cooling.

The major advantage of beryllium as a moderator is its (n,2n) reaction characteristic. Scattering along with the extra neutrons provided by this reaction give this system a high neutron economy. An outer reflector of water further aids in reducing absorption and leakage. Polyethylene plugs were the most effective moderator material for use in stainless steel targets because they overcome parasitic absorption by the iron better than beryllium plugs. Low absorption characteristics of zirconium allow extra neutrons to induce further fission events whereas the extra neutrons are lost to leakage and parasitic absorption with polyethylene plugs. Zirconium hydride plugs are generally less effective as plug materials for both types of target claddings.

Neutronic characteristics between beryllium and beryllium oxide are similar enough that beryllium oxide could be used if economic and material limitations prove the use of beryllium to be unfeasible. Material drawbacks with beryllium include its normally brittle nature and the production of helium gas which further enhances failure of the beryllium matrix with time and neutron irradiation. In addition, production of lithium-6 from the decay of helium-6 provides a significant amount of parasitic absorption which could affect long-term system criticality. Although beryllium oxide is easier to fabricate than beryllium, it vaporizes rapidly in the presence of moisture due to the formation of beryllium hydroxide. This could still pose a problem with using beryllium or its compounds as both are toxic when inhaled. Use of low enriched targets have been successfully illustrated here for beryllium moderated systems. However, their applicability for use in other moderators with this reactor concept is reserved for future research.

2.4.4 Graphite Moderator

The next moderator material tested was graphite. A total of ten configurations were modeled with graphite using stainless steel and zirconium targets. The target pitch, radial and axial reflector dimensions, and coolant channel size were optimized in a series of parametric studies. To reduce the neutron leakage outside the graphite reflectors, an additional water reflector 10 cm in thickness surrounds the exterior of the modeled reactors.

Table 5 presents the k_{eff} and standard deviation results obtained from MCNP for the graphite moderated reactors cooled with water. k_{eff} results for critical configurations appear in bold face type. Four types of solid plug materials for the target central region were modeled and compared to the base case of a dry center for stainless steel and zirconium targets. The target pitch obtained from the optimization studies for all modeled systems was 4.00 cm. Depending on the plug material in the target center, the target coolant channel size varied from 0.20 cm to 0.60 cm. All cases evaluated for graphite consisted of cores with 37 targets and a mass loading of 28.2 grams of ^{235}U per target.

Table 5. MCNP Results for Graphite Moderated Reactors

Target Pitch (cm)	Total Number of Targets*	²³⁵ U Mass (grams)	Cladding Material	Target Center	Water Gap (cm)	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	k _{eff} ± σ
4.00	37	28.2	SS	Dry	0.60	Graphite / H ₂ O	90.0 / 30.0 graphite 10.0 / 10.0 H ₂ O	0.75812 ± 0.00227
4.00	37	28.2	Zr	Dry	0.60	Graphite / H ₂ O	90.0 / 30.0 graphite 10.0 / 10.0 H ₂ O	0.90420 ± 0.00241
4.00	37	28.2	SS	Be	0.40	Graphite / H ₂ O	60.0 / 30.0 graphite 10.0 / 10.0 H ₂ O	0.88895 ± 0.00231
4.00	37	28.2	Zr	Be	0.40	Graphite / H ₂ O	60.0 / 30.0 graphite 10.0 / 10.0 H ₂ O	1.04688 ± 0.00300
4.00	37	28.2	SS	Graphite	0.60	Graphite / H ₂ O	80.0 / 30.0 graphite 10.0 / 10.0 H ₂ O	0.83256 ± 0.00251
4.00	37	28.2	Zr	Graphite	0.60	Graphite / H ₂ O	80.0 / 30.0 graphite 10.0 / 10.0 H ₂ O	0.98126 ± 0.00264
4.00	37	28.2	SS	Poly	0.20	Graphite / H ₂ O	80.0 / 40.0 graphite 10.0 / 10.0 H ₂ O	0.92140 ± 0.00323
4.00	37	28.2	Zr	Poly	0.20	Graphite / H ₂ O	80.0 / 50.0 graphite 10.0 / 10.0 H ₂ O	1.06668 ± 0.00347
4.00	37	28.2	SS	ZrH(1.85)	0.20	Graphite / H ₂ O	80.0 / 40.0 graphite 10.0 / 10.0 H ₂ O	0.88533 ± 0.00338
4.00	37	28.2	Zr	ZrH(1.85)	0.20	Graphite / H ₂ O	80.0 / 40.0 graphite 10.0 / 10.0 H ₂ O	1.02252 ± 0.00309

*All targets used 93% enriched uranium.

Based on the analysis of the graphite moderated reactors, a limited number critical configurations with 37 targets were discovered. From the ten graphite reactors evaluated, only three yielded a critical system. Moreover, these critical systems were limited to zirconium targets and required the use of a solid plug. The four solid plug materials tested were: graphite, beryllium, polyethylene, and zirconium hydride (64.9 % hydrogen). Figure 12 illustrates the k_{eff} results shown in Table 5 with the stainless steel targets. Figure 13 shows the k_{eff} results for zirconium targets.

The use of solid plugs did not aid in attaining a critical system with stainless steel targets. Each of the reactors composed of stainless steel targets had a k_{eff} of less than 0.95. The base case with a dry center is very subcritical with a k_{eff} which never exceeds 0.80. When graphite plugs are used, k_{eff} rises slightly but is still very subcritical with k_{eff} never exceeding 0.85. Zirconium hydride and beryllium plugs are fairly comparable but never exceed a k_{eff} of 0.90. The best plug material tested is polyethylene; however, the highest k_{eff} with this type of neutron scatterer never reaches 0.95.

Figure 12. k_{eff} Results for Graphite Moderated Reactors Using 37 Stainless Steel Clad Targets and Various Target Center Materials.

Figure 13. k_{eff} Results for Graphite Moderated Reactors Using 37 Zirconium Clad Targets and Various Target Center Materials.

There are a couple of reasons for the subcritical k_{eff} values observed with stainless steel targets in graphite. First is the parasitic absorption in the steel cladding, and second is the high leakage rate incurred as a result of neutron scattering by graphite and the plug materials. Target pitches over 4.00 cm were found to be in the over moderated regime in the pitch optimization study and would have yielded lower k_{eff} values than those shown in Figure 12. Figure 13 for the zirconium targets shows a dry target and the use of graphite plugs both yield subcritical systems and are considered poor design choices. As a result, polyethylene plugs are the moderator of choice for the target center plugs followed by beryllium and zirconium hydride plugs.

For both types of cladding materials, the optimized coolant channel thicknesses did not change; however, they were dependent upon the plug material. The optimum coolant channel size for dry targets and those with graphite plugs was 0.60 cm. These coolant channel sizes were larger than the others obtained in the study. A reduction to 0.40 cm of water with targets containing beryllium plugs was due to better scattering properties and the (n,2n) reactions. Polyethylene and zirconium hydride plugs required only 0.20 cm of water due to their rather high scattering properties and low absorption traits.

Graphite is generally a good moderator and reflector material. However, its ability to provide adequate neutron moderation and reflection with this reactor concept is limited. It fared poorly with stainless steel targets due to the high parasitic absorption of neutrons in the iron of stainless steel. In addition, the use of solid plug materials in the target center did provide a significant increase in k_{eff} . Consequently, there are two requirements for a target reactor to be constructed with a graphite moderator and reflector. First, the targets must be clad with zirconium metal. Second, a solid plug made of polyethylene, beryllium, or zirconium hydride must be used. The good scattering properties of these materials combined with the low absorption traits of zirconium provide a viable combination.

2.4.5 Heavy Water Moderator

The last moderator material investigated was heavy water. Four reactors with 37 targets in each core were constructed with stainless steel and zirconium targets. These reactors were modeled with two different coolant configurations:

- the first pair of reactors uses heavy water as the moderator and coolant; and
- the second pair uses water as the coolant and heavy water as the moderator.

There are two major advantages with using water as the coolant. It prevents costly and frequent replacement of heavy water boiled away due to heating. Also heavy water requires extreme pressurization if it is to be used as an effective coolant. Table 6 presents the results for the modeled heavy water moderated systems.

Table 6. MCNP Results for Heavy Water Moderated Reactors

Target Pitch (cm)	Total Number of Targets*	²³⁵ U Mass (grams)	Cladding Material	Target Center	Water Gap (cm)	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	$k_{eff} \pm \sigma$
11.50	37	28.2	SS	Dry	-	D ₂ O / H ₂ O	175.0 / 125.0 D ₂ O 10.0 / 10.0 H ₂ O	1.04686 ± 0.00276
11.50	37	28.2	Zr	Dry	-	D ₂ O / H ₂ O	175.0 / 125.0 D ₂ O 10.0 / 10.0 H ₂ O	1.21666 ± 0.00332
11.00	37	28.2	SS	Dry	0.20	D ₂ O / H ₂ O	150.0 / 150.0 D ₂ O 10.0 / 10.0 H ₂ O	1.00444 ± 0.00224
11.00	37	28.2	Zr	Dry	0.20	D ₂ O / H ₂ O	150.0 / 150.0 D ₂ O 10.0 / 10.0 H ₂ O	1.16557 ± 0.00282

*All targets used 93% enriched uranium.

The target pitch and dimensions of the radial and axial reflectors were determined using the same incremental parametric studies as the other moderator materials. The water coolant gap size was optimized simultaneously with the target pitch. An infinite reflector of water 10 cm surrounded the exterior of the reactor to reduce neutron leakage.

A major difference from the other moderator studies is that both pairs of reactors analyzed here do not use solid plugs in the central region of the target. The decision of whether or not to use solid plugs was determined in the target pitch optimization. Figure 14 illustrates the results for a reactor composed of 37 infinitely high targets, moderated and cooled with heavy water, and surrounded by an infinite heavy water reflector. The k_{∞} results for stainless steel and zirconium clad targets with a dry center are compared to those using beryllium or graphite plugs.

Figure 14. k_{∞} Results for 37 Targets in a Heavy Water Moderated Reactor with Various Target Pitches.

Figure 14 illustrates that the use of solid plugs for both cladding materials does not induce a significant rise in k_{∞} . The data shows there is virtually no difference in k_{∞} between graphite and beryllium plugs. This may be attributable to neutron leakage which is caused by the scattering properties of the plug materials and the heavy water. Consequently, solid plugs were not needed in the target central region with heavy water as the moderator.

Addition of a water coolant channel provides the advantage of preserving heavy water and minimizing expensive replacement. However, the addition of a water channel does induce a loss in reactivity when compared to a reactor where heavy water serves in the dual roles of moderator and coolant. For example, the loss in reactivity incurred for the reactors in Table 6 due to addition of a water coolant channel ranges from $-\$5.76 \pm 0.508$ with stainless steel targets to $-\$5.15 \pm 0.622$ for zirconium targets. Figure 15 is a plot showing the differences in k_{eff} results for a heavy water moderated system cooled with water or heavy water.

Figure 15. k_{eff} Results for 37 Targets in a Heavy Water Moderated Reactor Cooled with Heavy Water or Ordinary Water.

All of the reactors evaluated here yielded critical configurations with a k_{eff} value above 1.00. In comparison to water, all of the cores tested were able to attain criticality with 37 targets and without the use of solid plugs. This is primarily due to the scattering properties and low absorption cross section of heavy water. However, there are a couple of major disadvantages which limit selecting heavy water as the primary moderator for this reactor concept. First, the reactors analyzed have very large dimensions as compared to the smaller reactors moderated with other materials such as water. The large target pitch and immense quantities of heavy water required for the reflectors could make this type of reactor difficult and expensive to build. Furthermore, heavy water has a limited availability which renders it less attractive than other materials.

2.5 Conclusions

A number of trade-offs and important considerations are evident from the results of each moderator investigated in this optimization study. To attain a critical configuration in a water moderated system, targets would have to be manufactured with a low absorption material such as zirconium. In addition, the target center would require the use of a solid moderator material. The optimized reactor dimensions are fairly reasonable but a large number of targets are required to attain a critical system. Beryllium moderated systems yielded critical configurations with either cladding material and with a minimum of 37 targets. Most of these cases also required the use of solid plugs; however, a 37 target core of dry zirconium targets proved feasible.

Likewise, critical graphite moderated systems were possible with 37 targets, but required the use of plugs and zirconium cladding. System dimensions were fairly reasonable and comparable to water and beryllium systems. In contrast to the other moderators, heavy water moderated systems attained critical configurations with 37 targets and without the use of plugs. However, the dimensions of the systems were extremely large and impractical from a design and economic consideration. This study also indicated that low enriched targets can attain a critical mass in a beryllium core. To adequately assess their overall feasibility for use in full scale ^{99}Mo production, further work would be required.

3. TEMPERATURE & DENSITY EFFECTS ON k_{eff}

This chapter summarizes a series of parametric studies to assess the effects of temperature and density changes on k_{eff} . The optimized reactors described in Chapter 2 were analyzed for this study. A minimum of two configurations were chosen for each moderator in this investigation. The major parameters analyzed in this study were:

- moderator temperature;
- moderator density; and
- fuel temperature.

Moderator temperature and density effects were also evaluated for cases where a solid plug was used in the target central region. To obtain better statistical results, the number of particle histories was increased from the 100,000 used in the optimization study to 400,000. k_{eff} results for this study are summarized in terms of the reactivity losses incurred from a rise in temperature and a reduction in density. Incremental studies were done for each moderator to assess these effects on k_{eff} separately; however, the results presented here are for the combined effects and are considered conservative estimates for expected reactivity losses.

3.1 Hot and Cold Conditions

To adequately assess temperature and density effects, a set of temperatures and corresponding densities were defined for each material in the reactors tested at cold and hot conditions. Temperature changes were accounted for by using the $S(\alpha, \beta)$ cross sections in MCNP for the appropriate temperature and material. Table 7 lists the temperatures and densities used for various reactor materials at cold and hot conditions.

Temperatures for the materials in Table 7 were not chosen arbitrarily; rather, they were selected based on the available $S(\alpha, \beta)$ libraries in MCNP. For example, graphite and beryllium were evaluated at 600 K because an $S(\alpha, \beta)$ library does not exist for these materials at 400 K. Cases where water and heavy water were used as the coolant, hot conditions were defined with the same temperature and density as the moderator. Hot conditions for the secondary water reflector were defined at 300 K based on the assumption that any heat transferred from the targets to the moderator did not reach the outer reflector. Density changes in the beryllium moderator, beryllium and polyethylene plugs, and UO_2 fuel layer were not investigated assuming little change with temperature.

Table 7. Parameters for Various Reactor Materials at Cold & Hot Conditions.

Material	Cold Condition Temperatures (K)	Cold Condition Densities (g/cm ³)	Hot Condition Temperatures (K)	Hot Condition Densities (g/cm ³)
Water moderator & coolant	300	1.00	400	0.9431
Exterior water reflector	300	1.00	300	1.00
Graphite moderator	300	1.70	600	1.656
Beryllium moderator & plugs	300	1.85	600	1.85
Heavy water moderator & coolant	300	1.105	400	1.039
Polyethylene plugs	300	0.92	500	0.92
UO ₂ target layer	300	10.97	600	10.97

3.2 Water Moderator

Based on the reactor configurations investigated in the optimization study, only two reactors were deemed credible candidates for this analysis. These cores consisted of 85 targets, 30 grams of ²³⁵U per target, cooled with water, and containing beryllium plugs in the target central region. Table 8 summarizes the k_{eff} results at cold and hot conditions, and the resulting reactivity losses.

The results in Table 8 show both reactors with stainless steel and zirconium targets did not incur a large enough loss in reactivity to become subcritical at hot conditions. A smaller reactivity loss in zirconium targets is mainly attributable to the very low absorption cross section in zirconium. This allows for fission neutrons to be thermalized in the moderator and return to the fuel layer to induce further fission events; hence, a higher k_{eff} and lower reactivity loss.

Table 8. Results of Reactivity Losses for Water Moderated Systems.

Target Pitch (cm)	Total Number of Targets*	Target Clad	Target Center	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	Cold $k_{eff} \pm \sigma$	Hot $k_{eff} \pm \sigma$	Reactivity Loss $\Delta\rho$ (\$) ($\beta_{eff}=0.0070$)
3.95	85	SS	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.02621 $\pm$ 0.00130	1.00627 $\pm$ 0.00121	-2.76 $\pm$ 0.254
3.90	85	Zr	Be	H ₂ O / H ₂ O	10.0 / 10.0	1.17403 $\pm$ 0.00145	1.15391 $\pm$ 0.00147	-2.12 $\pm$ 0.295

*All targets used 93% enriched uranium with a target mass of 30.0 grams ²³⁵U.

3.3 Beryllium Moderator

Table 9 lists the k_{eff} results and resulting reactivity losses due to temperature and density changes for five reactor systems chosen from the optimization analysis. All of these reactors were composed of 37 targets, 28.2 grams of ^{235}U per target, moderated with beryllium, cooled with water, and in some cases containing solid plugs in the target center. Beryllium oxide moderated systems were not evaluated here due their similar neutronic properties with beryllium. It is assumed that comparable reactivity losses would be experienced with beryllium oxide.

Table 9 shows that zirconium targets with a dry center incurred the smallest reactivity loss of all configurations tested. Higher reactivity losses for zirconium targets with solid plugs may be the result of higher leakage rates due to the neutron scattering inherent with these plug materials.

Table 9. Results of Reactivity Losses for Beryllium Moderated Systems.

Target Pitch (cm)	Total Number of Targets*	Target Clad	Target Center	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	Cold $k_{\text{eff}} \pm \sigma$	Hot $k_{\text{eff}} \pm \sigma$	Reactivity Loss $\Delta\rho$ (\$) ($\beta_{\text{eff}}=0.0070$)
7.00	37	Zr	Dry	Be / H ₂ O	30.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.10985 $\pm$ 0.00147	1.10730 $\pm$ 0.00144	-0.296 $\pm$ 0.294
6.00	37	SS	Be	Be / H ₂ O	25.0 / 20.0 Be 10.0 / 10.0 H ₂ O	1.00296 $\pm$ 0.00123	0.99393 $\pm$ 0.00121	-1.29 $\pm$ 0.246
6.00	37	Zr	Be	Be / H ₂ O	25.0 / 25.0 Be 10.0 / 10.0 H ₂ O	1.19248 $\pm$ 0.00153	1.18042 $\pm$ 0.00158	-1.22 $\pm$ 0.314
6.00	37	SS	Poly	Be / H ₂ O	30.0 / 15.0 Be 10.0 / 10.0 H ₂ O	1.01372 $\pm$ 0.00110	1.00727 $\pm$ 0.00110	-0.902 $\pm$ 0.222
5.00	37	Zr	Poly	Be / H ₂ O	30.0 / 15.0 Be 10.0 / 10.0 H ₂ O	1.18267 $\pm$ 0.00116	1.17669 $\pm$ 0.00129	-0.614 $\pm$ 0.248

*All targets used 93% enriched uranium with a target mass of 28.2 grams ^{235}U , and a 0.20 cm gap for water cooling.

3.4 Graphite Moderator

Table 10 lists the k_{eff} results for two graphite moderated reactors evaluated at cold and hot conditions, and the resulting reactivity losses. The two configurations analyzed were composed of 37 zirconium targets, 28.2 grams of ^{235}U per target, moderated by graphite, cooled with water, and containing either polyethylene or beryllium plugs in the inner central region of the targets.

Table 10. Results of Reactivity Losses for Graphite Moderated Systems.

Target Pitch (cm)	Total Number of Targets*	Target Center	H ₂ O Gap (cm)	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	Cold $k_{\text{eff}} \pm \sigma$	Hot $k_{\text{eff}} \pm \sigma$	Reactivity Loss $\Delta\rho$ (\$) ($\beta_{\text{eff}}=0.0070$)
4.00	37	Be	0.40	graphite / H ₂ O	60.0 / 30.0 graphite 10.0 / 10.0 H ₂ O	1.04294 $\pm$ 0.00148	1.01243 $\pm$ 0.00145	-4.13 $\pm$ 0.296
4.00	37	Poly	0.20	graphite / H ₂ O	80.0 / 50.0 graphite 10.0 / 10.0 H ₂ O	1.06054 $\pm$ 0.00120	1.04677 $\pm$ 0.00123	-1.77 $\pm$ 0.245

*All targets used 93% enriched uranium with a target mass of 28.2 grams ^{235}U and zirconium clad.

Reactivity losses with these graphite moderated cores are much higher than the comparable beryllium moderated cores. This may be due to higher leakage rates with graphite as the moderator when compared to beryllium and water. Comparing the two plug material cases shows a dramatic difference. A higher reactivity loss may be incurred with beryllium plugs because the extra neutrons produced in (n,2n) reactions are lost to leakage and do not contribute to further fission events. On the other hand, scattering by polyethylene allows neutrons to be moderated in the water coolant channel and cause subsequent fission events.

3.5 Heavy Water Moderator

Two types of heavy water systems were investigated for temperature and density effects: one cooled with heavy water (no water gap), and the other cooled with ordinary water (0.20 cm water gap). The reactor cores listed in Table 11 consist of 37 targets with dry target centers each having a ^{235}U mass loading of 28.2 grams per target.

Table 11. Results of Reactivity Losses for Heavy Water Moderated Systems.

Target Pitch (cm)	Total Number of Targets*	Target Clad	H ₂ O Gap (cm)	Moderator / Reflector Material	Radial / Axial Reflector Thickness (cm)	Cold $k_{\text{eff}} \pm \sigma$	Hot $k_{\text{eff}} \pm \sigma$	Reactivity Loss $\Delta\rho$ (\$) ($\beta_{\text{eff}}=0.0070$)
11.50	37	SS	-	D ₂ O / H ₂ O	175.0 / 125.0 D ₂ O 10.0 / 10.0 H ₂ O	1.04480 $\pm$ 0.00128	0.97667 $\pm$ 0.00138	-9.54 $\pm$ 0.269
11.50	37	Zr	-	D ₂ O / H ₂ O	175.0 / 125.0 D ₂ O 10.0 / 10.0 H ₂ O	1.20983 $\pm$ 0.00153	1.12561 $\pm$ 0.00170	-8.83 $\pm$ 0.327
11.00	37	SS	0.20	D ₂ O / H ₂ O	150.0 / 150.0 D ₂ O 10.0 / 10.0 H ₂ O	1.00532 $\pm$ 0.00121	0.94831 $\pm$ 0.00121	-8.54 $\pm$ 0.244
11.00	37	Zr	0.20	D ₂ O / H ₂ O	150.0 / 150.0 D ₂ O 10.0 / 10.0 H ₂ O	1.16132 $\pm$ 0.00144	1.08673 $\pm$ 0.00161	-8.44 $\pm$ 0.309

*All targets used 93% enriched uranium with a target mass of 28.2 grams ^{235}U and a dry central region.

Of all the examined moderator materials, these heavy water moderated cores incurred the greatest amount of reactivity loss due to temperature and density effects. Heavy water moderated and cooled systems had reactivity losses of about \$1.00 more than heavy water moderated, water cooled reactors. Systems cooled and moderated by heavy water have higher leakage rates due to scattering. The water coolant channel outside the target allows for neutrons to be thermalized thereby having a greater probability of inducing further fission events instead of being absorbed or lost to leakage as is the case heavy water coolant.

3.6 Conclusions

Several important conclusions can be drawn from this analysis. Comparison of the reactivity losses among the four moderators investigated shows that heavy water moderated systems incurred the largest loss of reactivity due to changes in temperature and density. The moderator with second largest reactivity loss was water, followed by graphite, and beryllium with the smallest loss. The large reactivity loss with heavy water may be attributable to its scattering properties which induced a large amount of neutron leakage, thus a sharp reduction in k_{eff} . However, these losses in reactivity were not solely limited to contributions from the moderators.

Parasitic absorption in stainless steel cladding also proved to be another factor. In contrast to the strong absorption properties of the stainless steel, zirconium clad targets had smaller reactivity losses due to the low absorption properties of zirconium. Overall, a majority of the configurations (ten out of the thirteen) analyzed remained critical to be considered viable reactor designs despite changes in temperature and density.

4. FISSION PRODUCTS & NEUTRON POISONS

Chapter 4 summarizes the results of several parametric studies completed to determine the effects of fission products and other neutron poisons. Major fission products and poisons of concern in this analysis are: xenon-135 (^{135}Xe), samarium-149 (^{149}Sm), and lithium-6 (^6Li). ^{135}Xe and ^{149}Sm are products of the ^{235}U fission process and are formed in the UO_2 coating of the target. ^6Li is formed in beryllium via beta emission from ^6He . ^6He is the result of (n,α) reactions in the beryllium matrix. A brief summary of the equations used for determining the concentrations of ^6Li and ^3H are presented in Appendix B.3. A similar summary of the equations used for ^{135}Xe and ^{149}Sm are in Appendix B.4.

A select number of reactor combinations were chosen for this study from those analyzed for temperature and density effects. The three reactors chosen from the list of candidates analyzed in Chapter 3 consist of 37 targets and are considered to be representative cases from that list based on their physical attributes such as cladding material, moderator material, and plug material type. A second criterion in the selection process was to choose configurations with a reasonable k_{eff} margin which would be able to overcome the reactivity loss due to poisoning by ^{135}Xe , ^{149}Sm , and ^6Li . Water moderated cores were not chosen for two reasons: because of their low k_{eff} values; and because their optimum core size of 85 targets exceeds the baseline 37 target core which is deemed an efficient size for ^{99}Mo production. Heavy water moderated reactors were also eliminated from further consideration due to their extreme size even though their k_{eff} values were high as shown in the temperature and density effects analysis.

4.1 Background Information

^{135}Xe has a very large thermal absorption cross section of about 2.6×10^6 barns, and its concentration is dependent on the neutron flux level in the reactor; hence, the reactivity effects of ^{135}Xe are also time dependent. Figure 16 shows the decay chain and formation of ^{135}Xe from fission to its stable daughter ^{135}Ba .

Figure 16. Decay Chain and Nuclear Formation of ^{135}Xe .

During reactor operation, ^{135}Xe builds up to an equilibrium concentration which is determined by the decay rates of ^{135}Xe and ^{135}I . These decay rates compete during reactor shutdown due to the lack of ^{135}Xe absorption. The large concentration of ^{135}I present in the core decays and produces a buildup of ^{135}Xe . The decay of ^{135}I eventually equals that of ^{135}Xe causing a peak in the ^{135}Xe concentration. Subsequently, the decay rate of ^{135}I falls below that of ^{135}Xe producing a gradual reduction in the ^{135}Xe concentration. If the reactor is restarted before the previous equilibrium ^{135}Xe level is attained, neutron losses are accelerated by the renewed absorption of ^{135}Xe until the equilibrium level is reestablished. The primary aspects of ^{135}Xe formation considered here were the magnitude of its peak concentration, how soon this occurs after reactor shutdown, and the appropriate time to restart the reactor.

^{149}Sm has a thermal absorption cross section of about 40,000 barns, and is considered to be stable. Thus, it does not go away over time and has a constant poisoning effect. Figure 17 illustrates the formation of ^{149}Sm from fission.

Figure 17. Decay Chain and Nuclear Formation of ^{149}Sm .

In contrast to ^{135}Xe , the equilibrium concentration of ^{149}Sm does not depend on the reactor flux level and must be burned out of the reactor via absorption. During reactor operation, the concentration of ^{149}Sm builds up to an equilibrium level. This concentration builds to a higher level during shutdown due to the decay of ^{149}Pm and the absence of a neutron flux to provide burnout. After restart, the ^{149}Sm concentration decreases and eventually reestablishes the equilibrium concentration before shutdown. For the analyzed operating time of 7 days, the ^{149}Sm fraction of saturation and its corresponding mass in the core were parameters of consideration.

The reactors chosen also use either a beryllium moderator or solid beryllium plugs in the target central region. Consequently, consideration of ^6Li , helium gas, and tritium (^3H) formation in the beryllium matrix is important. Figure 18 illustrates the (n,2n) reaction in beryllium and the formation of two alpha particles or helium nuclei from ^8Be . Figure 19 shows the nuclear mechanisms required for the formation of ^6Li , ^6He , helium gas, and ^3H from beryllium. The ratio of (n,2n) reactions to (n, α) reactions is approximately 1.15.

Figure 18. ^9Be (n,2n) ^8Be Reaction.

Figure 19. Nuclear Formation of ^6Li , ^6He , and ^3H from Beryllium.

Two alpha particles or helium nuclei are formed in addition to those in the (n,2n) reaction. One helium nucleus is formed with ${}^6\text{He}$ in the beryllium matrix and the other is formed simultaneously with ${}^3\text{H}$. As a result of the reactions shown in Figures 18 and 19, a large amount of helium gas is formed in the beryllium matrix. Formation of helium gas in the beryllium matrix is important because it accelerates the structural degradation of beryllium which is naturally a brittle material. ${}^6\text{Li}$ has a fairly large thermal absorption cross section ($\sigma_a=945$ barns) and is stable. Its formation is the direct result of beta emission from ${}^6\text{He}$. ${}^6\text{Li}$ formation is important when considering reactivity loss due to its large absorption cross section. ${}^3\text{H}$ production was analyzed to determine its magnitude and how this affects long term waste management problems.

4.2 Reactivity Losses by ${}^{135}\text{Xe}$, ${}^{149}\text{Sm}$, and ${}^6\text{Li}$ Production

Table 12 presents the results for three configurations analyzed for reactivity loss due to the buildup of ${}^{135}\text{Xe}$, ${}^{149}\text{Sm}$ in the UO_2 coating inside each target, and ${}^6\text{Li}$ depending on whether a beryllium moderator or solid plug was used. The reactivity losses for ${}^{135}\text{Xe}$ and ${}^{149}\text{Sm}$ pertain to the amount of reactivity lost over the course of a 7-day operation period. Results in the last column of Table 12 are indicative of the expected reactivity loss caused by the formation of ${}^6\text{Li}$ over a 10-year time span. Ten years was chosen as the time scale for the analysis of ${}^6\text{Li}$ since it is expected that the beryllium moderator and plugs could be used for this amount of time without replacement. However, replacement time depends on the extent of helium gas formation. Exactly how helium gas production affects the structural integrity of the beryllium matrix is beyond the scope of this analysis and is not considered here.

Comparing the 37 stainless steel and zirconium clad target cores both containing polyethylene plugs shows a statistically insignificant increase in the reactivity loss due to ${}^{135}\text{Xe}$. A significant increase in reactivity loss with a graphite moderated system using beryllium plugs and zirconium targets is due to fission neutrons and some of the extra neutrons from (n,2n) reactions in beryllium being absorbed by ${}^{135}\text{Xe}$ in the UO_2 . Losses due to ${}^{149}\text{Sm}$ are statistically comparable for the three reactors and much lower than those for ${}^{135}\text{Xe}$.

Reactivity loss from the formation of ${}^6\text{Li}$ in the beryllium moderator is about 1.85 times higher with stainless steel than with zirconium targets. This is due to combined parasitic absorption effects from iron and ${}^6\text{Li}$. Losses are rather high when a beryllium plug is used as some of the neutrons interacting with the beryllium plug are parasitically absorbed by ${}^6\text{Li}$ instead of being used in further fission events. Consequently, this analysis shows that a core composed of zirconium clad targets without beryllium plugs has the lowest reactivity loss due to these three nuclides.

Table 12. Results of Reactivity Losses Due to ^{135}Xe , ^{149}Sm , and ^6Li .

Cladding Material	Target Center	Moderator / Reflector Material	Equilibrium ^{135}Xe Reactivity Loss $\Delta\rho$ (\$) ($\beta_{\text{eff}}=0.0070$)	Post-shutdown ^{135}Xe Reactivity Loss $\Delta\rho$ (\$) ($\beta_{\text{eff}}=0.0070$)	^{149}Sm Reactivity Loss $\Delta\rho$ (\$) ($\beta_{\text{eff}}=0.0070$)	^6Li Reactivity Loss $\Delta\rho$ (\$) ($\beta_{\text{eff}}=0.0070$)
SS	Poly	Be / H ₂ O	-3.52 ± 0.22	-5.24 ± 0.33	-0.29 ± 0.07	-0.52 ± 0.22
Zr	Poly	Be / H ₂ O	-3.48 ± 0.25	-5.18 ± 0.37	-0.28 ± 0.07	-0.28 ± 0.23
Zr	Be	Graphite / H ₂ O	-4.80 ± 0.28	-7.06 ± 0.41	-0.25 ± 0.14	-0.71 ± 0.29

A second aspect of ^{135}Xe formation was to determine when the peak ^{135}Xe concentration occurs after reactor shutdown and how soon after this peak can the reactor be restarted. The results in Table 12 show for the two beryllium moderated cores, reactivity loss due to the buildup of ^{135}Xe after shutdown amounts to an additional \$1.70 above what is lost due to equilibrium ^{135}Xe . For the graphite moderated core with beryllium plugs, an additional \$2.26 of reactivity is lost due to ^{135}Xe buildup after shutdown. The difference in these two reactivity losses shows an additional \$0.60 of reactivity is lost with a graphite moderated core. The following three plots illustrate profiles of the number of ^{135}Xe atoms present after reactor shutdown for a 7-day target irradiation time. Data for these plots was generated using a MathCAD program to calculate the post reactor shutdown ^{135}Xe concentration (Parma, 1998b). Each curve represents a profile of the number of ^{135}Xe atoms present over a decay time of 40 hours for a given target irradiation time in days.

Figure 20 is the post reactor shutdown ^{135}Xe for 37 stainless steel targets with polyethylene plugs in a beryllium moderated core. Figure 21 shows the post reactor shutdown ^{135}Xe for the same reactor consisting of zirconium clad targets. Figure 22 pertains to a graphite moderated core with zirconium clad targets containing beryllium plugs.

Figure 20. Post Reactor Shutdown ^{135}Xe for 37 SS Targets with Polyethylene Plugs in a Beryllium Moderator.

Figure 21. Post Reactor Shutdown ^{135}Xe for 37 Zr Targets with Polyethylene Plugs in a Beryllium Moderator.

Figure 22. Post Reactor Shutdown ^{135}Xe for 37 Zr Targets with Beryllium Plugs in a Graphite Moderator.

Figures 20, 21, and 22 show several important features. First, the peak ^{135}Xe concentration occurs around 7.5 hours after shutdown for each of the reactors investigated. This is less than the typical peak in a light water reactor which occurs around 9 hours after shutdown. Second, the number of ^{135}Xe atoms present in the reactor after shutdown reaches saturation levels when the targets have been irradiated for about 2.5 days. Third, these plots also show that the equilibrium ^{135}Xe concentration is reestablished about 20 hours after shutdown. The only noticeable difference between the three reactors is that the equilibrium ^{135}Xe concentration is slightly higher for a graphite moderated core compared to a beryllium moderated system. Based on the k_{eff} values presented in Tables 9 and 10 for these three reactors and their corresponding reactivity losses due to ^{135}Xe in Table 12, only a core with zirconium targets and polyethylene plugs in a beryllium moderator has a sufficient k_{eff} margin to withstand a large reactivity loss.

Table 13 presents the results for the amount of ^{149}Sm formed in the core at saturation ($t=\infty$) and at the proposed operation time of 7 days. The fraction of saturation at 7 days is approximately 29%, with full saturation occurring at roughly 90 days. Results show that very low amounts of ^{149}Sm , on the order of 0.05 grams, are formed at saturation. Even lesser amounts, on the order of 0.016 grams, are formed at the proposed operation time of 7 days.

This is mainly due to the uranium content in each target. In comparison, typical fuel elements composed of disks or pellets contain much more uranium and ultimately yield more ^{149}Sm . These low amounts of ^{149}Sm are also consistent with the low reactivity losses shown in Table 12 for the three reactors.

Table 13. ^{149}Sm Mass and Fraction of Saturation Results for 37 Target Cores.

Cladding Material	Target Center	Moderator / Reflector Material	^{149}Sm Mass in Fuel (grams at $t=\infty$)	^{149}Sm Mass in Fuel (grams at $t=7$ days)	^{149}Sm Fraction of Saturation (%)
SS	Poly	Be / H_2O	0.0530	0.0156	29.39
Zr	Poly	Be / H_2O	0.0540	0.0158	29.22
Zr	Be	Graphite / H_2O	0.0540	0.0157	29.05

4.3 ^6Li , Helium Gas, and ^3H Production

Table 14 presents the results for ^6Li , helium gas, and ^3H production in the beryllium matrix of the three reactors analyzed. Formation of these nuclides was analyzed for a 10-year time span assuming 10 years is the lifetime of the beryllium moderator or plugs. Overall, ^6Li , ^3H , and helium gas concentrations are slightly higher for a core composed of stainless steel targets moderated by beryllium than a core with zirconium clad targets. In contrast, quantities of these nuclides in the graphite moderated core with beryllium plugs are lower than the beryllium moderated reactors for the same time scale of 10 years. This is primarily because there is much less beryllium for neutrons to interact with to produce ^6Li . In general, the activity of ^3H generated at the end of 10 years is high and could pose a problem with disposition of the beryllium moderator.

The number of helium gas atoms for each reactor in Table 14 represents all of the atoms produced through $(n,2n)$ and (n,α) reactions with beryllium in addition to helium atoms produced from (n,α) reactions with ^6Li to form ^3H (See Appendix B.2). Further complicating matters, the large quantity of helium gas produced accelerates the embrittlement and structural degradation of the beryllium matrix. If severe embrittlement due to helium gas production and neutron irradiation occurs, this could pose a problem with disposition of beryllium due to its toxicity when in a powder form.

Results of some preliminary calculations are presented in the last column of Table 14 which illustrate the ratio of beryllium atoms relative to the number of helium gas atoms produced. These numbers show that there are numerous beryllium atoms surrounding each helium atom in the moderator surrounding each stainless steel or zirconium target. Therefore, it could be assumed that the He gas does not diffuse out but is locked in the matrix of the beryllium moderator. In contrast for a beryllium plug, there are only 1,158 atoms surrounding each helium gas atom produced. For this case, it would be conservative to assume there might be some leakage of helium from the matrix of a beryllium plug. Nevertheless, further calculations would have to be done to verify these assumptions.

Table 14. ^6Li , ^3H , and He Concentrations for 37 Target Cores

Cladding Material	Target Center Material	Moderator / Reflector Material	Beryllium Volume (cm^3)	^6Li Atoms (at $t=10$ yrs)	^3H Activity (Ci at $t=10$ yrs)	He Gas Atoms (at $t=10$ yrs)	$N_{\text{Be}}/N_{\text{He}}$ (atoms Be/atom He)
SS	Poly	Be / H_2O	5.48×10^5	2.47×10^{23}	2.44×10^4	3.25×10^{24}	20,857
Zr	Poly	Be / H_2O	4.79×10^5	2.14×10^{23}	2.05×10^4	2.75×10^{24}	21,529
Zr	Be	Graphite / H_2O	1.07×10^4	5.41×10^{22}	9.23×10^3	1.15×10^{24}	1,158

4.4 Conclusions

Several important findings were discovered concerning the effects of fission products and neutron poisons in this analysis. Total reactivity losses due to ^{135}Xe at the peak concentration after shutdown varied from around \$5.15 to nearly \$7.00 for the three reactors analyzed here. Consequently, overcoming reactivity losses due to ^{135}Xe production could play a big role in determining reactor shutdown scheduling and control schemes. Only the beryllium moderated core with zirconium targets had sufficient excess reactivity to overcome a peak in the ^{135}Xe concentration after reactor shutdown. Reactivity effects due to ^{149}Sm production were small in the three reactors, with the fraction of saturation near 30% at the end of the proposed operation time of 7 days. Production of ^6Li in the beryllium moderator or plugs was small compared to the large helium gas volumes and ^3H activities produced over the same time period of 10 years. Further analysis would have to be done to determine if significant volumes of helium gas and large ^3H activities could reduce the utilization time and adversely affect waste disposition of the beryllium moderator or plugs.

5. FUEL BURNUP

Chapter 5 presents and explains the results of a parametric analysis designed to quantify the reactivity effects of ^{235}U burnup in the UO_2 coating of the target. The three reactors analyzed previously in Chapter 4 are evaluated for ^{235}U burnup effects. The major objective of this study is to confirm that fuel burnup is minimal and does not impose a large loss in reactivity for the proposed operation time of 7 days. Positive reactivity effects due to the production of ^{239}Pu from (n,γ) reactions with ^{238}U in the UO_2 are not considered. A majority of the uranium in each target consists of ^{235}U and it is expected that very little ^{239}Pu is produced.

5.1 Fuel Burnup Results

Each of the three reactors consist of 37 targets loaded with 28.2 grams of ^{235}U per target. To obtain reasonable statistics, it was assumed fuel burnup would be uniform across the entire core of targets. A reasonable rate for the amount of ^{235}U fissioned in each target was postulated to be 10% of the original ^{235}U loading in each target. Therefore, a 10% burnup rate translates to 2.82 grams of ^{235}U per target or 104.34 grams of ^{235}U in the entire core being fissioned or burned. This rate of burnup is a conservative estimate as the entire core will be refueled with new targets every 7 days. Reactivity loss estimates only account for a reduction in ^{235}U per target due to burnup and do not include fission product effects. Table 15 presents the projected reactivity losses per gram of ^{235}U fissioned for the three target reactors.

Results in Table 15 do not exceed 10 cents per gram of ^{235}U burned for any of the three reactors evaluated. At the end of a 7 day operation time, only 5.72 grams of ^{235}U is burned in the entire core. For example, this translates to a total reactivity loss of \$0.32 for the first reactor in Table 15. Consequently, this demonstrates the 10% rate is a conservative prediction for all reactors in Table 15 as only 0.55% of the entire inventory of ^{235}U is fissioned at 7 days.

Table 15. Reactivity Losses Due to 10% ^{235}U Burnup for 37 Target Cores

Total Number of Targets	^{235}U Mass (grams)	^{235}U Mass at 10% Burnup (grams)	Cladding Material	Target Center	Moderator / Reflector Material	Reactivity Loss with 10% ^{235}U Burnup $\Delta\rho$ (\$/gram ^{235}U) ($\beta_{\text{eff}}=0.0070$)
37	28.2	25.38	SS	Poly	Be / H_2O	-0.056 ± 0.0021
37	28.2	25.38	Zr	Poly	Be / H_2O	-0.032 ± 0.0023
37	28.2	25.38	Zr	Be	Graphite / H_2O	-0.029 ± 0.0028

The beryllium moderated core with stainless steel targets has the largest amount of reactivity loss followed by the same beryllium moderated reactor with zirconium targets. Irradiation of stainless steel targets incurs the largest amount of reactivity loss due to the combined contributions of parasitic absorption in the steel and a loss in fissile material. Reactivity loss in a graphite moderated core with zirconium targets is fairly comparable to its beryllium counterpart.

5.2 Conclusions

This analysis clearly shows that a fuel burnup rate of 10% does not present a significant loss in reactivity for the three reactors investigated. Reactivity losses did not exceed 10 cents per gram of ^{235}U fissioned, with the highest estimate near 6 cents per gram of ^{235}U and the lowest near 3 cents per gram of ^{235}U . As a result, a 10% burnup rate can be considered a conservative estimate of reactivity loss for these reactors based on the amount of ^{235}U loaded in each target and the proposed irradiation time of 7 days.

6. ECONOMIC ANALYSIS

Chapter 6 presents and analyzes the results of a two-phase economic study designed to provide approximate costs for the initial construction of the proposed invention; and waste generation expenses for the conventional, the Cintichem-type target fueled, and the solution reactor concepts. Analysis of accelerator produced ^{99}Mo is not considered because estimates of these expenses are not currently available. Two major objectives of this analysis are:

1. to demonstrate that a target-fueled reactor can be constructed at a reasonable cost; and
2. to show significant savings can be incurred with a target-fueled reactor by the elimination of a driver core of fuel elements.

This investigation is not meant to provide an in-depth economic analysis; rather, the intention is to give estimates for the basis of making a reasonable and sound comparison of the production concepts.

Many of the target-fueled configurations modeled and optimized with MCNP are neutronically viable; however, it was necessary to perform this analysis to quantify the economic ramifications of this new production concept. The first phase of this analysis consisted of determining initial construction estimates for three 37 target cores chosen from the list of candidates in Chapter 3. Two of the three reactors are beryllium moderated, water cooled, and contain polyethylene plugs in the center of the targets. These two configurations are directly comparable; the only difference being they both have different cladding materials. The third reactor is graphite moderated, water-cooled, and contains zirconium clad targets with beryllium plugs in the central region.

Table 16 presents cost estimates for initial construction of these reactors based on the optimized dimensions determined with MCNP as presented in Chapter 1. A sample hand calculation of how the cost estimates were derived is provided in Appendix B.1. Water moderated reactors were excluded from this analysis as the sizes of these cores are much larger than the baseline configuration of 37 targets. It is assumed that the cost of building a water moderated reactor would be comparable to that of a graphite moderated system. Estimates of the initial construction costs for a heavy water moderated reactor were provided to illustrate that even though a 37 target core moderated with this material is neutronically viable, it is obviously very expensive and economically unfeasible.

Results from the second phase of the analysis are outlined in Table 17. This table is a summary of the annual cost estimates for waste generation expenses envisioned for the conventional and solution reactor methods as compared to those for the proposed reactor concept. Original estimates of annual high level waste (HLW) disposal assume 57 elements per year with 16 kg U/yr (US DOE/EIS-0249D, Table 5-19, 1995) for the ACRR. This amount of uranium pertains to the current $\text{UO}_2\text{-BeO}$ fuel which will not be used for production purposes. HLW disposal estimates for the ACRR are based on the new production fuel which will contain approximately 5 kg of UO_2 and 1 kg of stainless steel clad per fuel element. Low level waste (LLW) volumes for a conventional reactor pertain to those expected for the ACRR and HCF at 100% production capacity (US DOE/EIS-0249D, Table 5-19, 1995). Cost and waste volume estimates in the last column of Table 17 reflect previous calculations for ^{99}Mo produced via a solution reactor (Glenn et al., Tables I & II, 1997). Unit prices in the second column of Table 17 were not used to calculate costs of LLW and liquid waste disposal for the solution reactor.

Table 16. Initial Construction Cost Estimates of 37 Target Cores.

	Unit Price	Beryllium Moderated Reactor	Graphite Moderated Reactor	Heavy Water Moderated Reactor
Number of Targets	37 targets \$1,000 / target	37 targets \$37,000	37 targets \$37,000	37 targets \$37,000
Moderator Cost	\$1,775 / lb Be ^a \$198 / lb graphite ^b \$654.65 / kg D ₂ O ^c	2,236 lb Be \$3,970,000	7,211 lb graphite \$1,428,000	57,355 kg D ₂ O \$37,550,000
Plug Material	\$1,775 / lb Be	Polyethylene 37 plugs	Beryllium 37 plugs \$78,000	No plugs \$0
Total Cost (\$)		≈\$4,004,000	≈\$1,543,000	≈\$37,600,000

Table 17. Comparison of Annual Costs for Various ⁹⁹Mo Production Methods.

	Unit Price	Conventional Reactor (ACRR)	Cintichem-type Target Fueled Reactor	Solution Reactor
Target Fabrication	\$1,000/target ^d	1924 targets \$1,924,000	1924 targets \$1,924,000	0 targets \$0
Annual Fuel Costs	\$20,000/fuel element ^e \$6,300/kg solution fuel ^f	57 fuel elements \$1,114,000	—	0.06 kg ^g \$380
HLW Disposal				
Spent Nuclear Fuel	\$100/kg ^h	342 kg/yr (57 elements) \$34,200	0 m ³ \$0	0 m ³ \$0
LLW Disposal				
Routine	\$353/m ³ ⁱ	15.74 m ³ \$5,550	15.74 m ³ \$5,550	25.5 m ³ \$8,900
Solidified	\$353/m ³	42.0 m ³ \$14,800	42.0 m ³ \$14,800	0.06 m ³ \$6
Ion Exchange Resins	\$353/m ³	1.70 m ³ \$600	1.70 m ³ \$600	3.40 m ³ \$1200
Targets	\$353/m ³	7.0 m ³ \$2,470	7.0 m ³ \$2,470	0 targets \$0
Extraction Columns	\$353/m ³	0 columns \$0	0 columns \$0	8.00 m ³ \$2,850
Liquid Waste Disposal				
Routine	\$353/m ³	7.3 m ³ \$2,580	7.3 m ³ \$2,580	8.00 m ³ \$2,000
Target Fabrication	\$353/m ³	100 m ³ ^j \$35,300	100 m ³ \$35,300	0 targets \$0
Total Cost (\$)		\$3,159,500	\$1,985,300	\$15,386

^a Personal communication with Rebecca Brown, Electronic Space Products International, August 1998.

^b Personal communication with Rebecca Brown, Electronic Space Products International, August 1998.

^c URL: <http://www.wilmad.com/html/nf/DeuteriumOxide.html>

^d Personal Communication with E.J. Parma, SNL, June 1998.

^e Personal Communication with E.J. Parma, SNL, June 1998.

^f Glenn, D.E., MS Thesis, University of New Mexico, Appendix A.13, 1995.

^g Glenn, D.E., MS Thesis, University of New Mexico, p. 35, 1995

^h Glenn, D.E., MS Thesis, University of New Mexico, Table 3.1, 1995.

ⁱ Personal Communication with S.W. Longley, SNL, September 1998.

^j Massey and Coats, SAND95-2703, Table A-4

The total cost estimates for initial construction shown in Table 16 reveal a couple of important observations, the first of which is the moderator cost. Expense incurred for an initial loading of targets is considered to be equivalent among the three reactors. A heavy water moderated core proved to be the least desirable moderator material investigated here due to its large construction cost, which is a direct function of the reactor size. On the other hand, beryllium is relatively less expensive compared to heavy water and could be substituted as a reasonable choice. Even though the core size of a graphite moderated reactor is similar to a beryllium moderated system, graphite is the least expensive material of the group.

When selecting the optimum moderator material, size and associated expense should not be the only factors driving a decision. A second and more important observation evident from these results is the balanced trade-off that must be made between these economic costs and the neutronic properties. Heavy water in general demonstrated good neutronic properties with high k_{eff} values, its only major disadvantage were the large reactivity losses incurred due to temperature and density changes. Coupling this to its exorbitant cost and limited distribution in the United States make it a poor choice overall.

Beryllium is a very good moderator due to its scattering properties and its ability to produce (n,2n) reactions. Its reactivity losses due to temperature and density changes, ^{149}Sm buildup, ^6Li production, and 10% ^{235}U burnup are low. However, three issues that could prove a beryllium moderated core to be a poor choice are: major reactivity losses due to ^{135}Xe buildup; and the production of large quantities of helium gas and ^3H . Graphite is a good moderator; however, its scattering properties lend it to producing more neutron leakage compared to beryllium as shown by the lower k_{eff} values. Reactivity losses due to temperature and density effects are very high, nearly \$3.00 above those for beryllium. Likewise, losses for ^{135}Xe buildup after shutdown are nearly \$2.00 higher than those for beryllium. Figures for ^{149}Sm , ^6Li , and 10% ^{235}U burnup are low and comparable to beryllium.

Based on the evidence presented, the beryllium moderated system is better neutronicallly but more expensive. A graphite moderated system is the exact opposite, less expensive but less desirable neutronicallly. Thus, combining the above trade-offs yields two reactors which could be considered optimal final choices. These are reactors fueled with zirconium clad targets in a beryllium or graphite moderator. Unfortunately, one major drawback to these reactors would be the need to use zirconium cladding. Neutronicallly, zirconium is a very good material due to low neutron absorption; however, its use would probably require a redesign of the target and possibly a new UO_2 coating process as the current electroplating process is unproven with zirconium targets.

Several assumptions were made with regard to calculating the projected costs in Table 17. First, target fabrication costs were considered to be comparable for the ACRR and target reactor concept. Therefore, these are accounted for only as a fuel cost for the target reactor. Second, fuel costs for the solution reactor were obtained from calculations for 20% enriched solution fuel (Glenn, Appendix A.13, 1995). Third, HLW for the scope of this analysis is classified as spent fuel in solid or liquid form depending on the reactor type. Current HCF waste disposition plans call for shipping processed target shells with the solidified liquid waste from chemical extraction operations. Consequently, HLW is not accounted for with the target reactor because

based on this assumption, the target shells would be treated as LLW. Likewise, Glenn assumes in his calculations that the solution reactor fuel will not be considered HLW based on the assumption that Ci concentrations for LLW classification will not be exceeded (Glenn et al., 1997).

A second aspect of this analysis was to assess the economic feasibility of the target reactor concept compared to conventional and solution reactors. A comparison of the costs associated with eliminating the driver core of fuel elements incurs a savings of approximately \$1.14 million. This is assuming the costs of purchasing targets for irradiation with both reactors over the course of one operation year are equivalent. Much less fuel is required for a solution reactor and is shown to be less expensive than the reactor methods. However, a thorough feasibility study has not been done to assess the core lifetime of a solution reactor based on fission product buildup, temperature effects on reactivity, etc. If results of such a study would prove that more fuel is required to overcome these effects, then it is expected the overall cost of such a reactor system (especially fuel) will increase significantly. HLW disposal costs are eliminated with a target-fueled reactor due to the lack of a driver core of fuel elements and the assumption that the targets will be treated as LLW. HLW disposal costs with the solution reactor are also eliminated based on the same assumption.

LLW disposal costs are considered to be equivalent between the conventional and target reactor methods. This is because the Cintichem process is the only method that will be used to chemically extract ^{99}Mo from the targets. In addition, LLW disposal costs of solidified liquid waste from chemical extraction operations and general liquid waste from target fabrication are much higher than for the solution reactor due to the utilization of targets. Large scale tests to date have not been conducted with the solution reactor to determine its feasibility for a commercial scale production of ^{99}Mo comparable to the conventional reactor method. Consequently, solidified liquid and HLW costs documented here would rise if more fuel was deemed necessary to meet the U.S. demand for ^{99}Mo with the solution reactor. There are no costs associated with the use of extraction columns for the reactors, but a large volume and cost must be accounted for with the solution reactors.

6.1 Conclusions

The results of this economic analysis illustrate the conventional reactor method is the most expensive of the three production methods. Likewise, the cost estimates and assumptions for the target reactor clearly confirm that the new reactor concept is a significant improvement over the conventional reactor. A major enhancement in terms of economics is the elimination of HLW disposal costs and the notable financial savings incurred from elimination of the driver core of fuel elements. These are two aspects with this reactor concept which could render it a competitive option in the future.

Beryllium and graphite proved to be the least expensive of the four moderators investigated. Balancing the economic impacts of constructing the reactors chosen for this study and their neutronic requirements discovered from analysis presented in previous chapters yielded two credible candidates. One requires the use of a beryllium moderator and the other a graphite moderator; however, both also require the target cladding to be composed of zirconium. Zirconium cladding works very well neutronically, but the economic impacts of its potential

have not been realized with respect to the current coating process which could possibly entail a complete overhaul of the current target design. Furthermore, an comprehensive and complete analysis cannot be achieved for the comparison of ^{99}Mo production methods until estimates for accelerator produced ^{99}Mo are detailed in the areas of construction costs and waste generation.

7. CONCLUDING REMARKS

7.1 Recommendations For Future Work

Several areas of this project are recommended for consideration as future work. Among the first of these is a thermal-hydraulics analysis for the two configurations chosen as final candidates. Pursuing such an analysis for these two and possibly other configurations analyzed is considered beyond the scope of this endeavor. This type of analysis would require further optimization studies to ensure the neutronic requirements of a particular reactor system are consistent with its thermal-hydraulic limitations. A second recommendation for future research is further conceptual development and parametric studies with LEU coated targets. Results from a limited parametric study done here with LEU targets moderated with beryllium clearly show there is potential for their use in a target-fueled reactor. Optimization studies with other moderators would need to be done for further validation.

A major impediment to the use of LEU targets is lack of authorization from the FDA for a Master Drug File approving the production and processing of fission produced ^{99}Mo from LEU targets, and there are no immediate plans to do so. Two major reasons are the reluctance of pharmaceutical companies to accept fission produced ^{99}Mo with LEU because of radionuclidic impurity issues, and utilization of the Cintichem process with this type of uranium is unproven. Implementation of control rods into the MCNP model was not done nor were parametric studies conducted in this investigation. Parametric analyses to optimize a control system are essential to quantifying how much excess reactivity would be required to overcome reactivity losses from the largest contributors determined in this study, namely losses due to temperature and density changes in the moderator, and fission product poisoning by ^{135}Xe .

Further studies should be considered to examine the impacts of material degradation with the solid moderators of beryllium and graphite. Lining the inside of the coolant channels would need to be done with a material that does not compromise the system neutronics (a material other than stainless steel), but also prevents accelerated material degradation by the interaction of water with the beryllium or graphite. Results showed significant quantities of He gas are produced in beryllium moderated systems which adversely affect its structural integrity; however, encasement of the beryllium could prove to be a viable solution in simplifying waste disposition and reducing human exposure to beryllium powder. Encasement of the beryllium could also prove to be a feasible solution to containment of large quantities of ^3H produced in the beryllium matrix.

It is conceivable that stainless steel targets could still be used in a final reactor design if the current process for electroplating UO_2 on zirconium cladding is not possible without seriously compromising the Cintichem process. A system with the nominal target design could be constructed with the addition of moderator plugs in the central region of the targets, or by increasing the core size slightly beyond 37 targets to more efficiently overcome large reactivity losses. For example, moderator plugs such as polyethylene are so inexpensive to manufacture that they could easily be disposed of after each irradiation cycle. Some of these major issues require further research to complete a comprehensive and thorough design of this reactor concept. These and other questions not mentioned here would have to be resolved before any experimental studies could be done with a prototype of this reactor.

7.2 Final Conclusions

The primary objective of this feasibility investigation was to determine if a critical reactor composed solely of Cintichem-type targets could be constructed without the aid of a driver core while retaining the capability of supplying the U.S. demand for ^{99}Mo and preserving the Cintichem process. The following lists highlight the major conclusions obtained from numerous parametric studies conducted in this analysis. Many feasible combinations were obtained from optimization studies with various moderators and target core sizes which showed:

- zirconium was the optimum cladding material due to its low neutron absorption characteristics, but would probably require a redesign of the current target if used;
- adding solid moderator plugs to the target central regions aided in attaining criticality with subcritical systems;
- water systems had fairly reasonable sizes but required a minimum of 85 targets with solid plugs of beryllium to reach criticality;
- beryllium systems had reasonable sizes, low neutron leakage and high k_{eff} values due to $(n,2n)$ reactions, and yielded critical systems with stainless steel and LEU targets;
- graphite systems had comparable sizes to beryllium, but critical systems were limited to zirconium clad targets with solid plugs; and
- D_2O systems had high k_{eff} values without the aid of solid plugs, but had very large core sizes.

The field of potential candidates for additional analysis was narrowed to thirteen configurations. Further parametric studies were done with these thirteen to assess reactivity losses due to temperature and density effects. Results from these studies showed:

- temperature and density effects were dependent on moderator material;
- a decrease in k_{eff} occurred with an increase in moderator temperature and a decrease in moderator density; and
- beryllium moderated systems yielded the smallest reactivity loss while D_2O moderated systems suffered the largest reactivity loss.

To obtain two configurations as final candidates, this pool of candidates was further narrowed from thirteen to three for supplementary parametric studies. Reactivity effects by fission products such as ^{135}Xe and ^{149}Sm and neutron poisoning by ^6Li production were investigated and showed:

- post reactor shutdown reactivity losses due to ^{135}Xe were highest for a graphite system and about \$2.00 less with a comparable system of zirconium or stainless steel clad targets in beryllium;
- reactivity losses due to ^6Li and ^{149}Sm production were small;
- helium gas production was largest in the beryllium moderated cores with less production in the graphite system with beryllium plugs; and
- ^3H activities were high in the beryllium moderated systems.

Reactivity losses due to fuel burnup at a rate of 10% in the three configurations were found to be very small.

An economic analysis was included to determine initial construction cost estimates for the proposed concept, and to provide a comparison of the projected annual waste disposal costs for conventional, target, and solution reactor methods. Results of the economic analysis showed:

- initial construction costs for a graphite moderated core are relatively inexpensive, with a D₂O moderated core being the most expensive;
- elimination of the driver core saves about \$1.14 million with no associated HLW disposal costs, and LLW disposal costs are comparable between the conventional and target reactors; and
- even though previous calculations of waste disposal costs for a solution reactor method are lower when compared to a conventional reactor, additional and large scale tests are necessary to fully assess the validity of these economic savings (i.e. LLW waste costs could rise if more fuel is need to meet U.S. ⁹⁹Mo demand).

Finally, the first of two configurations which prevailed as final candidates is a beryllium moderated system cooled with water, and composed of 37 zirconium clad targets with polyethylene plugs. The second configuration is moderated by graphite, cooled with water, and contains beryllium plugs in 37 zirconium clad targets. These two are very diverse in terms of their physical characteristics as well as their neutronic and economic limitations. Furthermore, the results obtained for these systems definitely confirm that this reactor concept could demonstrate to be a viable and competitive production option which is capable of supplying the entire U.S. demand for ⁹⁹Mo while ultimately preserving the Cintichem process.

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APPENDIX A

COMMONLY USED ABBREVIATIONS & ACRONYMS

ACCR	Annular Core Research Reactor (Sandia National Laboratories, NM)
AECL	Atomic Energy of Canada, Ltd.
ATR	Advanced Test Reactor (Idaho National Engineering Lab)
B&W	Babcock & Wilcox Environmental Services, Inc.
Ci	Curie
DOE	United States Department of Energy
FDA	United States Food and Drug Administration
GETR	General Electric Test Reactor
HCF	Hot Cell Facility (Sandia National Laboratories, NM)
HEU	highly enriched uranium
HFIR	High Flux Isotope Reactor (Oak Ridge National Laboratory)
HLW	high level waste
INEEL	Idaho National Engineering & Environmental Laboratory
IPDP	Isotope Production & Distribution Program
LAMPF	Los Alamos Meson Physics Facility
LANL	Los Alamos National Laboratory
LEU	low enriched uranium
LLW	low level waste
MIPR	Medical Isotope Production Reactor
MURR	Missouri University Research Reactor (Columbia, Missouri)
NRC	United States Nuclear Regulatory Commission
NRU	National Research Universal Reactor (Canada)
ORNL	Oak Ridge National Laboratory
ROD	Record of Decision
SNL/NM	Sandia National Laboratories, NM
SNM	Special Nuclear Material
UCD	University of California-Davis
k_{eff}	effective multiplication factor
k_{∞}	infinite multiplication factor
^3H	hydrogen-3 (tritium)
^6He	helium-6
^6Li	lithium-6
^9Be	beryllium-9
^{96}Tc	technetium-96
^{99}Mo	molybdenum-99
^{99}Ru	ruthenium-99
$^{99\text{m}}\text{Tc}$	metastable technetium-99
^{135}Xe	xenon-135
^{149}Sm	samarium-149
^{235}U	uranium-235
^{238}U	uranium-238
^{239}Pu	plutonium-239
β_{eff}	effective delayed neutron fraction

APPENDIX B

SAMPLE HAND CALCULATIONS

B.1: Reactor Construction Cost Estimate

The following calculation details how the initial construction cost estimates were calculated for the configurations listed in Table 16, Chapter 6. This example is for a graphite moderated system composed of 37 zirconium clad targets with beryllium plugs in the center of each target.

Given parameters: 37 zirconium clad targets
74 beryllium slugs (2 per target)
Graphite moderator
Beryllium plugs
\$198.00 / lb graphite
\$1775.00 / lb beryllium
\$1000 / target
Axial core height = 108.26 cm
Radius of core = 75.0 cm
Coolant channel radius = 1.90 cm
Coolant channel height = 108.26 cm
Be slug radius = 1.50 cm
Be slug height = 30.0 cm
Beryllium density = 1.85 g/cm³
Graphite density = 1.70 g/cm³

The total volume of the reactor core with the given dimensions is,

$$V_{core} = \pi R_{core}^2 H_{core} = \pi (75.0 \text{ cm})^2 (108.26 \text{ cm}) = 1.913 \times 10^6 \text{ cm}^3$$

Above and below each target is a solid slug of moderator material to stabilize the target in the coolant channel and further reduce neutron leakage in the axial direction. The volume of one beryllium slug is,

$$V_{slug} = \pi R_{slug}^2 H_{slug} = \pi (1.50 \text{ cm})^2 (108.26 \text{ cm}) = 765.24 \text{ cm}^3$$

The total volume of all the beryllium slugs in a core consisting of 37 targets is,

$$V_{slugs}^{total} = \left(2 \frac{\text{slugs}}{\text{target}} \right) (37 \text{ targets}) \left(212.05 \frac{\text{cm}^3}{\text{slug}} \right) = 5.663 \times 10^4 \text{ cm}^3$$

To provide coolant to the targets, coolant channels are drilled into the core lattice. The volume of one target coolant channel is,

$$V_{channel} = \pi R_{channel}^2 H_{channel} = \pi (1.90cm)^2 (108.26cm) = 1227.29cm^3$$

Total volume of the target coolant channels in a core of 37 targets is,

$$V_{channels}^{total} = (37 channels) \left(1227.29 \frac{cm^3}{channel} \right) = 4.541 \times 10^4 cm^3$$

Subtracting the total volume of the coolant channels from the total volume of the core gives the true volume of graphite in the core.

$$V_{core} = (1.913 \times 10^6 - 4.541 \times 10^4) cm^3 = 1.867 \times 10^6 cm^3$$

Adding the total volume of the slugs above and below each target to the true volume of graphite in the core gives,

$$V_{core \& slugs} = (1.867 \times 10^6 + 5.663 \times 10^4) cm^3 = 1.924 \times 10^6 cm^3$$

To find the mass of graphite in the core based on the volume,

$$m_{graphite} = \rho * V = \left(1.70 \frac{g}{cm^3} \right) (1.924 \times 10^6 cm^3) = 3.271 \times 10^6 g = 3,271 kg$$

$$m_{graphite} = (3,271 kg) \left(2.2046 \frac{lb}{kg} \right) = 7,211 lb$$

The cost of the graphite moderator in the core based on the given unit price is,

$$\text{Unit cost of graphite} = \$198.00 / lb$$

Based on this unit price for graphite, the total cost of the graphite moderator is,

$$\text{Total cost of graphite} = (7211 lb) (\$198.00 / lb) = \$1,427,778 \approx \$1,428,000$$

To maintain criticality, the central region of each target contains a solid cylindrical plug of a moderator material. In this case, the volume of a beryllium plug in the target center is,

$$V_{Be plug} = \pi R_{Be plug}^2 H_{Be plug} = \pi (1.4208cm)^2 (45.72cm) = 289.95cm^3$$

And the mass of one beryllium plug is,

$$m_{Be} = \rho * V = \left(1.85 \frac{g}{cm^3}\right) (289.95 cm^3) = 536.41g$$

To convert the mass of one beryllium plug into pounds,

$$m_{Be} = (536.41g) \left(\frac{2.2046 \times 10^{-3} lb}{g} \right) = 1.183lb$$

The unit price of beryllium and the resulting cost of the beryllium based on the mass of each plug required for 37 targets is,

$$\text{Unit cost of beryllium} = \$1775.00 / lb$$

$$\text{Total cost of beryllium plugs} = \left(\frac{\$1775.00}{lb} \right) (37 \text{ plugs}) \left(\frac{1.183 lb}{\text{plug}} \right)$$

$$\text{Total cost of beryllium plugs} = \$77,664.69 \cong \$78,000$$

The approximate cost of one stainless steel Cintichem-type target is \$1000. Because zirconium targets are used in this configuration and the cost of such a target is unknown, it is assumed to be that of a stainless steel target for this calculation. Thus, for a 37 target core,

$$\text{Cost of one target} = \left(\frac{\$1000.00}{\text{target}} \right) (37 \text{ targets}) = \$37,000$$

Adding the costs of the beryllium plugs, graphite core, and targets finally gives,

$$\text{Total Cost} = \$37,000 + \$78,000 + \$1,428,000$$

$$\therefore \text{Total Cost} \cong \$1,543,000$$

B.2: Helium Gas Content Calculation

The helium gas production was determined using a cell flux tally in MCNP. This type of tally counts the number of particles interacting with a particular cell volume containing a specific material. MCNP normalizes this tally to the number of particles crossing the volume. To obtain the number of He atoms produced as a result of (n,2n) and (n,α) reactions in beryllium, a cell flux tally was implemented over the beryllium moderator in the core or solid plugs in the targets for the analyzed configurations listed in Table 14, Chapter 4. The following sample calculation summarizes how the magnitude of helium gas production was obtained for a configuration consisting of 37 stainless steel targets, moderated with beryllium, cooled by water, and containing polyethylene plugs in the central region of each target.

For the (n,2n) reactions in beryllium, two helium nuclei are produced (from the decay of a ^8Be atom) every time a fission neutron interacts with a beryllium atom.

To obtain the number of (n,2n) reactions that occur, a unit analysis must be done with the MCNP cell tally $\sigma_{\text{Be}}\phi_t$ for beryllium. Multiplying the tally result by the tallied volume of $5.66464 \times 10^5 \text{ cm}^3$ gives,

$$\sigma_{\text{Be}}\phi_t * V_{\text{tally}} = \left(6.78398 \times 10^{-7} \frac{\text{rxns}}{\text{cm}^3 * n_s * \frac{\text{atom}}{b - \text{cm}}} \right) (5.66464 \times 10^5 \text{ cm}^3) = 0.384288 \frac{\text{rxns}}{n_s * \frac{\text{atom}}{b - \text{cm}}}$$

The actual volume of the beryllium core excluding the coolant channels was calculated to be $548,199.58 \text{ cm}^3$. Using this number to correct the tally for the actual volume over which the reactions occur, and using the corresponding number of fissions occurring per source neutron gives,

$$\begin{aligned} \sigma_{\text{Be}}\phi_t &= \left(0.384288 \frac{\text{rxns}}{n_s * \frac{\text{atom}}{b - \text{cm}}} \right) \left(\frac{1}{548199.58 \text{ cm}^3} \right) \left(\frac{10^{-24} \text{ cm}^2}{b} \right) \left(\frac{1}{0.4154307 \frac{\text{fiss}}{n_s}} \right) \\ \sigma_{\text{Be}}\phi_t &= 1.687405 \times 10^{-30} \frac{\text{rxns}}{\text{atomBe}^* \text{ fiss}} \end{aligned}$$

To account for reactor power, it is necessary to multiply the tally by the amount of power in MW*sec per fission, assuming 200 MeV per fission.

$$\left(200 \frac{\text{MeV}}{\text{fiss}}\right) \left(\frac{1.602 \times 10^{-13} \text{ J}}{\text{MeV}}\right) \left(\frac{\text{W}}{\text{J/sec}}\right) \left(\frac{\text{MW}}{10^6 \text{ W}}\right) = 3.204 \times 10^{-17} \frac{\text{MW} * \text{sec}}{\text{fiss}}$$

In units of fissions/MW*sec, the previous figure is 3.1211×10^{16} fissions/MW*sec. Using this conversion factor to adjust the Be tally $\sigma_{\text{Be}}\phi_t$ for reactor power,

$$\sigma_{\text{Be}}\phi_t = \left(1.687405 \times 10^{-30} \frac{\text{rxns}}{\text{atomBe} * \text{fiss}}\right) \left(\frac{3.1211 \times 10^{16} \text{ fiss}}{\text{MW} * \text{sec}}\right)$$

$$\sigma_{\text{Be}}\phi_t = 5.26655 \times 10^{-14} \frac{\text{rxns}}{\text{atomBe} * \text{MW} * \text{sec}}$$

Now, we need to calculate the number of beryllium atoms in the moderator matrix.

$$N_{\text{Be}} = \frac{\rho * N_A}{A} = \left(1.85 \frac{\text{gBe}}{\text{cm}^3}\right) \left(\frac{\text{molBe}}{9.012 \text{ gBe}}\right) \left(\frac{6.022 \times 10^{23} \text{ atomsBe}}{\text{molBe}}\right) = 1.236 \times 10^{23} \frac{\text{atomsBe}}{\text{cm}^3}$$

$$n_{\text{Be}} = N_{\text{Be}} * V_{\text{Be}} = \left(1.236 \times 10^{23} \frac{\text{atomsBe}}{\text{cm}^3}\right) (548199.58 \text{ cm}^3) = 6.7757 \times 10^{28} \text{ atomsBe}$$

Using the tally result $\sigma_{\text{Be}}\phi_t$ which was adjusted for reactor power and the calculated number of beryllium atoms in the moderator, we find the percentage of (n,2n) reactions that occur as a result of fission neutrons interacting with a Be nucleus,

$$\frac{\#(n,2n) \text{ rxns}}{\text{fiss}} = \left(1.687405 \times 10^{-30} \frac{\text{rxns}}{\text{atomBe} * \text{fiss}}\right) (6.7757 \times 10^{28} \text{ atomsBe})$$

$$\frac{\#(n,2n) \text{ rxns}}{\text{fiss}} = 0.1143 \frac{\text{rxns}}{\text{fiss}} = 11.43\%$$

Assuming the 37 targets are operating at 21 kW each and the moderator will be used for 10 years, the amount of energy over this time period is,

$$E = (37 \text{ targets}) \left(21 \frac{\text{kW}}{\text{target}}\right) \left(\frac{\text{MW}}{10^3 \text{ kW}}\right) (3.1536 \times 10^8 \text{ sec}) = 2.4503 \times 10^8 \text{ MW} * \text{sec}$$

Converting this amount of power to the corresponding number of fissions, this is:

$$\# \text{ fissions} = (2.4503 \times 10^8 \text{ MW} \cdot \text{sec}) \left(3.1211 \times 10^{16} \frac{\text{fiss}}{\text{MW} \cdot \text{sec}} \right) = 7.6476 \times 10^{24} \text{ fissions}$$

Since there are 2 helium nuclei for every (n,2n) reaction, then total number of helium nuclei produced is,

$$n_{\text{He}}(n,2n) = \left(2 \frac{\text{atomsHe}}{\text{rxn}} \right) \left(0.1143 \frac{\text{rxns}}{\text{fiss}} \right) (7.6476 \times 10^{24} \text{ fiss}) = 1.7482 \times 10^{24} \text{ atomsHe}$$

There are also two other helium nuclei which are formed in the beryllium matrix as a result of (n, α) reactions in beryllium to produce ${}^6\text{Li}$, and (n, α) reactions with ${}^6\text{Li}$ to produce ${}^3\text{H}$.

For the first set of (n, α) reactions, the production rate of ${}^6\text{Li}$ is,

$$P = (\sigma_{\text{Be}} \phi_t) * N_{\text{Be}}$$

$$P = \left(5.2665 \times 10^{-14} \frac{\text{rxns}}{\text{atomBe} \cdot \text{MW} \cdot \text{sec}} \right) (6.7757 \times 10^{28} \text{ atomsBe}) = 3.56845 \times 10^{15} \frac{\text{rxns}}{\text{MW} \cdot \text{sec}}$$

Assuming that the moderator will be used for a 10 year period, the integrated number of ${}^6\text{Li}$ nuclei produced from the β^- decay of ${}^6\text{He}$ in the first set of (n, α) reactions at 10 years is,

$${}^6\text{Li}_{\text{tot}} = \left(3.56845 \times 10^{15} \frac{\text{rxns}}{\text{MW} \cdot \text{sec}} \right) (2.4503 \times 10^8 \text{ MW} \cdot \text{sec}) = 8.7437 \times 10^{23} \text{ atoms } {}^6\text{Li} \& \text{ He}$$

To obtain the equilibrium ${}^6\text{Li}$ concentration, the cell tally $\sigma_{\text{Li}} \phi_t$ for ${}^6\text{Li}$ formation in the beryllium is used in this calculation. Multiplying the tally result by the tallied volume,

$$\sigma_{\text{Li}} \phi_t * V_{\text{tally}} = \left(1.79751 \times 10^{-1} \frac{\text{rxns}}{\text{cm}^3 * n_s * \frac{\text{atom}}{b - \text{cm}}} \right) (5.66464 \times 10^5 \text{ cm}^3) = 101822.47 \frac{\text{rxns}}{n_s * \frac{\text{atom}}{b - \text{cm}}}$$

Correcting the cell tally $\sigma_{Li}\phi_t$ for the actual volume over which the (n, α) reactions occur gives,

$$\sigma_{Li}\phi_t = \left(10182247 \frac{rxns}{n_s * \frac{atom}{b-cm}} \right) \left(\frac{1}{54819958 cm^3} \right) \left(\frac{10^{-24} cm^2}{b} \right) \left(\frac{1}{0.4154307 \frac{fiss}{n_s}} \right)$$

$$\sigma_{Li}\phi_t = 4.47101 \times 10^{-25} \frac{rxns}{atomLi * fiss}$$

Converting the above result to account for reactor power,

$$\sigma_{Li}\phi_t = \left(4.47101 \times 10^{-25} \frac{rxns}{atomLi * fiss} \right) \left(\frac{3.121 \times 10^6 fiss}{MW * sec} \right) = 1.39544 \times 10^{-8} \frac{rxns}{atomLi * MW * sec}$$

The equilibrium concentration of 6Li is found by dividing the 6Li production rate into the cell tally $\sigma_{Li}\phi_t$,

$$Li(\infty) = \frac{P}{\sigma_a^{Li} \phi_t} = \frac{3.56845 \times 10^{15} \frac{rxns}{MW * sec}}{1.39544 \times 10^{-8} \frac{rxns}{atomLi * MW * sec}} = 2.5572 \times 10^{23} atoms {}^6Li$$

Adjusting this result to the time frame of 10 years as a fraction of the saturated concentration,

$$Li(10 yrs) = \frac{P}{\sigma_a^{Li} \phi_t} \left[1 - e^{-\sigma_a^{Li} \phi_t t} \right]$$

$$Li(10 yrs) = \left(2.5572 \times 10^{23} atoms {}^6Li \right) \left[1 - e^{-\left(1.39544 \times 10^{-8} \right) \left(3650 d \right) \left(\frac{24 hr}{d} \right) \left(\frac{3600 s}{hr} \right) (0.777 MW)} \right]$$

$$Li(10 yrs) = 2.4734 \times 10^{23} atoms {}^6Li = 0.9672 Li(\infty)$$

So the resulting number of (n, α) reactions with beryllium is,

$$\frac{\#(n, \alpha)_{Be} rxns}{fiss} = \left(4.47101 \times 10^{-25} \frac{rxns}{atomLi * fiss} \right) (2.5572 \times 10^{23} atoms Li)$$

$$\frac{\#(n, \alpha)_{Be} rxns}{fiss} = 0.1143 \frac{rxns}{fiss} = 11.43\%$$

For the second set of helium atoms produced by (n,α) reactions with ${}^6\text{Li}$, there is one helium nucleus formed with every ${}^3\text{H}$ atom. To obtain this, we must subtract the number of ${}^6\text{Li}$ atoms as a fraction of saturation from the integrated number of ${}^6\text{Li}$ atoms formed at 10 years,

$${}^6\text{Li}_{\text{tot}} - {}^6\text{Li}(10\text{yrs}) = (8.7437 \times 10^{23} - 2.4734 \times 10^{23}) = 6.2703 \times 10^{23} \text{ atomsHe} \& {}^3\text{H}$$

or

$$n_{\text{He}}(n, \alpha) = \frac{6.2703 \times 10^{23} \text{ atomsHe}}{7.6476 \times 10^{24} \text{ fissions}} = 0.08199 \frac{\text{atomsHe}}{\text{fiss}}$$

Therefore, the total number of helium atoms as a result of (n,2n) and (n,α) reactions is,

$$n_{\text{He}}^{\text{tot}} = \left((0.2286 + 0.1143 + 0.08199) \frac{\text{atomsHe}}{\text{fiss}} \right) (7.6476 \times 10^{24} \text{ fissions})$$

$$\therefore n_{\text{He}}^{\text{tot}} = 3.2487 \times 10^{24} \text{ atomsHe}$$

Using the molar gas conversion at STP, this corresponds to a volume of,

$$V_{\text{He}} = (3.2487 \times 10^{24} \text{ atomsHe}) \left(\frac{\text{molHe}}{6.022 \times 10^{23} \text{ atoms}} \right) \left(\frac{22.4\text{L gas}}{\text{molHe}} \right) = 120.8\text{L He gas}$$

To preliminarily quantify if the beryllium matrix will hold the helium gas formed, the ratio of beryllium atoms relative to the number of helium atoms is,

$$\frac{N_{\text{Be}}}{N_{\text{He}}} = \left(\frac{6.7757 \times 10^{28} \text{ atomsBe}}{3.2487 \times 10^{24} \text{ atomsHe}} \right) = 20,857 \frac{\text{atoms Be}}{\text{atom He gas}}$$

Therefore, it is conservative to assume that with this many beryllium atoms surrounding every helium atom produced, the beryllium matrix should retain the helium gas over a 10 year period.

B.3: ${}^6\text{Li}$ and ${}^3\text{H}$ Analysis

The following differential equations describe the ${}^6\text{Li}$ and ${}^3\text{H}$ concentrations produced in the beryllium moderator of the core or the plugs in the central region of a target as a function of time.

$$\frac{dLi}{dt} = P - \sigma_a^{Li} \phi_t Li$$

$$\frac{dT}{dt} = \sigma_a^{Li} \phi_t Li - \lambda_T T$$

The variables Li and T are the ${}^6\text{Li}$ and ${}^3\text{H}$ concentrations, P is the production rate of ${}^6\text{Li}$ from (n, α) reactions with beryllium, and λ is the decay constant. The $\sigma_a \phi_t$ term was obtained from particle tallies with MCNP across the moderator surrounding each target or the beryllium plug in the central region of a target. Solving the above differential equations by integration to obtain the number of ${}^6\text{Li}$ and ${}^3\text{H}$ atoms at time t yields the following,

$$Li(t) = \frac{P}{\sigma_a^{Li} \phi_t} \left[1 - e^{-\sigma_a^{Li} \phi_t t} \right]$$

$$T(t) = \frac{P}{\lambda_T} \left[1 + \frac{1}{\lambda_T - \sigma_a^{Li} \phi_t} \left(\sigma_a^{Li} \phi_t e^{-\lambda_T t} - \lambda_T e^{-\sigma_a^{Li} \phi_t t} \right) \right]$$

At time $t = \infty$ which represents equilibrium or saturation conditions, the above solutions reduce down to the following,

$$Li(\infty) = \frac{P}{\sigma_a^{Li} \phi_t}$$

$$T(\infty) = \frac{P}{\lambda_T}$$

To convert the ${}^3\text{H}$ concentrations to units of activity at time t and $t = \infty$, the following equation was used,

$$A = -\lambda N$$

where A is the ${}^3\text{H}$ activity at time t in Ci, λ is the decay constant in sec^{-1} , and N is the number of ${}^3\text{H}$ atoms at time t .

B.4: ^{135}Xe and ^{149}Sm Fission Product Analysis

The following equations were used to quantify the concentrations of ^{135}I and ^{135}Xe produced in the fuel from fission as a function of time.

$$\frac{dI}{dt} = \gamma_I \Sigma_f \phi_t V - \lambda_I I$$

$$\frac{dX}{dt} = \lambda_I I + \gamma_X \Sigma_f \phi_t V - \lambda_X X - \sigma_a^X \phi_t X$$

The variables X and I are the ^{135}Xe and ^{135}I concentrations, λ is the decay constant, γ is the fission yield of ^{135}Xe and ^{135}I , V is the fuel volume, and Σ_f is the macroscopic fission cross section of the fuel. The $\sigma_a \phi_t$ was obtained from MCNP using particle tallies across the fuel region of each target. Solving the above equations by integration gives the following concentrations or number of ^{135}I and ^{135}Xe atoms at time t as:

$$I(t) = \frac{\gamma_I P [1 - e^{-\lambda_I t}]}{\lambda_I}$$

$$X(t) = \frac{\gamma_X P}{\lambda_X + \sigma_a^X \phi_t} - \frac{\gamma_X P e^{-(\lambda_X + \sigma_a^X \phi_t)t}}{\lambda_X + \sigma_a^X \phi_t} + \lambda_I \gamma_I P \left[\frac{1}{\lambda_I} * \frac{1}{\lambda_X + \sigma_a^X \phi_t} - \frac{1}{\lambda_I} * \frac{e^{-\lambda_I t}}{\lambda_X + \sigma_a^X \phi_t - \lambda_I} + \frac{e^{-\lambda_I t}}{\lambda_X + \sigma_a^X \phi_t - \lambda_I} * \frac{1}{\lambda_X + \sigma_a^X \phi_t - \lambda_I} \right]$$

where P is the reactor power in MW. At time $t=\infty$, which represents equilibrium concentration conditions, the above equations reduce to,

$$I(\infty) = \frac{\gamma_I P}{\lambda_I}$$

$$X(\infty) = \frac{(\gamma_I + \gamma_X) P}{\lambda_I + \sigma_a^X \phi_t}$$

The following equation describes the ^{135}Xe concentration in the reactor after shutdown. This equation was employed in the MathCAD program (Parma, 1998b) to determine the magnitude of and the time at which the ^{135}Xe concentration peak occurs after reactor shutdown for several target irradiation times.

$$X(t) = X_o e^{-\lambda_x t} + \frac{\lambda_I I_o}{\lambda_I - \lambda_x} (e^{-\lambda_x t} - e^{-\lambda_I t})$$

where X_o and I_o are the equilibrium ^{135}Xe and ^{135}I concentrations. To convert the number ^{135}Xe and ^{135}I atoms at a given time t and at $t=\infty$ to units of activity in Ci, the following equation was used,

$$A = -\lambda N$$

where N is the number of ^{135}Xe or ^{135}I atoms at time t , λ is the decay constant in sec^{-1} , and A is the nuclide activity at time t in Ci.

The following equations define the concentrations of ^{149}Pm and ^{149}Sm present in the fuel layer of each target at time t ,

$$\frac{dPm}{dt} = \gamma_{Pm} \Sigma_f \phi_t V - \lambda_{Pm} Pm$$

$$\frac{dS}{dt} = \lambda_{Pm} Pm - \sigma_a^S S \phi_t$$

where Pm and S are the ^{149}Pm and ^{149}Sm concentrations, λ is the decay constant, γ is the fission yield of ^{149}Pm and ^{149}Sm , V is the total fuel volume, and Σ_f is the macroscopic cross section of the fuel. Solving these equations by integration gives the number of ^{149}Pm and ^{149}Sm atoms present as the following,

$$Pm(t) = \frac{\gamma_{Pm} P [1 - e^{-\lambda_{Pm} t}]}{\lambda_{Pm}}$$

$$S(t) = \frac{\lambda_{Pm} P}{\sigma_a^S \phi_t} - \frac{\lambda_{Pm} P}{(\sigma_a^S \phi_t - \lambda_{Pm})} \left[e^{-\lambda_{Pm} t} - \frac{\lambda_{Pm}}{\sigma_a^S \phi_t} e^{-\sigma_a^S \phi_t t} \right]$$

where P is the reactor power in MW. The $\sigma_a \phi_t$ was obtained from MCNP using particle tallies across the fuel region of each target.

At time $t=\infty$ which represents saturation or equilibrium concentration conditions, the previous equations become,

$$Pm(\infty) = \frac{\gamma_{Pm} P}{\lambda_{Pm}}$$

$$S(\infty) = \frac{\lambda_{Pm} P}{\sigma_a^S \phi_t}$$

To convert the ^{149}Sm concentration at a given time t and at $t=\infty$ to units of mass in grams, the following equation was used,

$$m_{Sm} = \frac{N_{Sm} * A_{Sm}}{N_A}$$

where N_{Sm} is the number of ^{149}Sm atoms, N_A is Avogadro's number, and A_{Sm} is the atomic weight of ^{149}Sm .

APPENDIX C

SAMPLE MCNP INPUT & OUTPUT FILE

C.1: MCNP 4A Sample Input File

```
multiple Cintichem targets, k-effective simulation
c
c =====
c 28.2 grams 93% U235 in UO2
c triangular target pitch
c target cladding thickness 0.0762 cm
c target diameter 3.00 cm
c 37 targets
c =====
c
c
c CELL CARDS
c
c inner void of target
c 1 0 -800 700 -3 u=1 imp:n=1
c polyethylene plug in inner void
1 6 0.11849 -800 700 -3 u=1 imp:n=1
c ZrH(1.85) plug in inner void
c 1 7 0.1034 -800 700 -3 u=1 imp:n=1
c
c 28.2 grams U235 in target layer
2 1 0.190316 -800 700 3 -2 u=1 imp:n=1
c
c zirconium target clad
c 3 2 -6.5 -801 701 -1 (2:800:-700) u=1 imp:n=1
c
c SS316 target clad
3 3 -8.03 -801 701 -1 (2:800:-700) u=1 imp:n=1
c
c top Be slug
4 5 -1.85 -1 801 u=1 imp:n=1
c
c bottom Be slug
5 5 -1.85 -1 -701 u=1 imp:n=1
c
c water gap outside target
6 4 -1.0 1 -4 u=1 imp:n=1
c 6 5 -1.85 1 -4 u=1 imp:n=1 $no water gap
c
c beryllium moderator
7 5 -1.85 4 u=1 imp:n=1
c
c
c inner void of target
c 11 0 -800 700 -3 u=3 imp:n=1
c polyethylene plug in inner void
11 6 0.11849 -800 700 -3 u=3 imp:n=1
c ZrH(1.85) plug in inner void
c 11 7 0.1034 -800 700 -3 u=3 imp:n=1
c
c 28.2 grams U235 in target layer
12 1 0.190316 -800 700 3 -2 u=3 imp:n=1
c
c zirconium target clad
c 13 2 -6.5 -801 701 -1 (2:800:-700) u=3 imp:n=1
c
c SS316 target clad
13 3 -8.03 -801 701 -1 (2:800:-700) u=3 imp:n=1
c
c top Be slug
14 5 -1.85 -1 801 u=3 imp:n=1
c
```

```

c bottom Be slug
15 5 -1.85 -1 -701 u=3 imp:n=1
c
c water gap outside target
16 4 -1.0 1 -4 u=3 imp:n=1
c 16 5 -1.85 1 -4 u=3 imp:n=1 $no water gap
c
c beryllium moderator
17 5 -1.85 4 u=3 imp:n=1
c
c
c inner void of target
c 21 0 -800 700 -3 u=4 imp:n=1
c polyethylene plug in inner void
21 6 0.11849 -800 700 -3 u=4 imp:n=1
c ZrH(1.85) plug in inner void
c 21 7 0.1034 -800 700 -3 u=4 imp:n=1
c
c 28.2 grams U235 in target layer
22 1 0.190316 -800 700 3 -2 u=4 imp:n=1
c
c zirconium target clad
c 23 2 -6.5 -801 701 -1 (2:800:-700) u=4 imp:n=1
c
c SS316 target clad
23 3 -8.03 -801 701 -1 (2:800:-700) u=4 imp:n=1
c
c top Be slug
24 5 -1.85 -1 801 u=4 imp:n=1
c
c bottom Be slug
25 5 -1.85 -1 -701 u=4 imp:n=1
c
c water gap outside target
26 4 -1.0 1 -4 u=4 imp:n=1
c 26 5 -1.85 1 -4 u=4 imp:n=1 $no water gap
c
c beryllium moderator
27 5 -1.85 4 u=4 imp:n=1
c
c
c inner void of target
c 31 0 -800 700 -3 u=5 imp:n=1
c polyethylene plug in inner void
31 6 0.11849 -800 700 -3 u=5 imp:n=1
c ZrH(1.85) plug in inner void
c 31 7 0.1034 -800 700 -3 u=5 imp:n=1
c
c 28.2 grams U235 in target layer
32 1 0.190316 -800 700 3 -2 u=5 imp:n=1
c
c zirconium target clad
c 33 2 -6.5 -801 701 -1 (2:800:-700) u=5 imp:n=1
c
c SS316 target clad
33 3 -8.03 -801 701 -1 (2:800:-700) u=5 imp:n=1
c
c top Be slug
34 5 -1.85 -1 801 u=5 imp:n=1
c
c bottom Be slug
35 5 -1.85 -1 -701 u=5 imp:n=1
c
c water gap outside target
36 4 -1.0 1 -4 u=5 imp:n=1
c 36 5 -1.85 1 -4 u=5 imp:n=1 $no water gap
c
c beryllium moderator
37 5 -1.85 4 u=5 imp:n=1
c
c

```



```

c axial and radial water reflector
10 8 -1.00 -1000 -803 703 (900:802:-702) imp:n=1
c
c outside world
c reflected reactor
110 0 1000:803:-703 imp:n=0
c bare reactor
c 10 0 (100:-200:300:-400:500:-600:800:-700) imp:n=0

c
c SURFACE CARDS
c
1 cz 1.50 $outer radius of target
2 cz 1.4238 $inner radius of target clad
3 cz 1.4208 $inner radius of U235 layer
4 cz 1.70 $outer radius of water gap
c 4 cz 1.90
c 4 cz 2.10
c 4 cz 2.30
c 4 cz 2.50
c
c small hexagons for targets in lattice
c
c p=5.00 cm
c 11 px 2.50
c 12 px -2.50
c 13 p 0.577350269 1.0 0.0 2.88675134
c 14 p 0.577350269 1.0 0.0 -2.88675134
c 15 p -0.577350269 1.0 0.0 2.88675134
c 16 p -0.577350269 1.0 0.0 -2.88675134
c
c p=6.00 cm
11 px 3.00
12 px -3.00
13 p 0.577350269 1.0 0.0 3.46410161
14 p 0.577350269 1.0 0.0 -3.46410161
15 p -0.577350269 1.0 0.0 3.46410161
16 p -0.577350269 1.0 0.0 -3.46410161
c
c
c
700 pz -22.86 $bottom of target (height=18 inches or 45.72 cm)
701 pz -24.13 $bottom cap on target
702 pz -39.13 $bottom of beryllium axial reflector
703 pz -49.13 $bottom of water reflector
800 pz 22.86 $top of target
801 pz 24.13 $top cap on target
802 pz 39.13 $top of beryllium axial reflector
803 pz 49.13 $top of water reflector
c
c p=5.00 cm (Zr clad)
c 900 cz 45.00
c 1000 cz 55.00
c
c p=6.00 cm (SS clad)
900 cz 48.00
1000 cz 58.00

c
c MATERIAL CARDS
c
mode n
kcode 1000 1.0 10 410
ksrc 1.4225 0.0 0.0
1.4225 0.0 4.00
1.4225 0.0 -4.00
1.4225 0.0 6.00
1.4225 0.0 -6.00
c
c
c

```

```

c UO2 layer in target
m1      92235.50c    0.05905    $uranium-235
        92238.50c    0.00439    $uranium-238
        8016.50c    0.12688    $oxygen
c
c Zirconium target cladding (6.5 g/cc)
m2      40000.50c    1.00        $zirconium
c
c Stainless steel target cladding (8.03 g/cc)
m3      24000.50c    -0.1853    $chromium
        25055.50c    -0.017     $manganese-55
        26000.50c    -0.6884    $iron
        28000.50c    -0.1034    $nickel
        14000.50c    -0.0059    $silicon
c
c Water moderator (1.00 g/cc)
m4      1001.50c    2.00        $hydrogen
        8016.50c    1.00        $oxygen
mt4     lwtr.01t    $thermal s(a,b) identifier 300K
c
c Beryllium moderator (1.85 g/cc)
m5      4009.50c    1.00        $beryllium
mt5     be.01t     $thermal s(a,b) identifier 300K
c
c Polyethylene plug (0.92 g/cc)
m6      6000.50c    0.03950    $carbon
        1001.50c    0.07899    $hydrogen
mt6     poly.01t   $thermal s(a,b) identifier 300K
c
c Zirconium hydride plug (1.85 H in Zr)
m7      40000.50c    0.03629    $zirconium
        1001.50c    0.06715    $hydrogen
mt7     h/zr.01t   $thermal s(a,b) identifier 300K
c
c Water reflector (1.00 g/cc)
m8      1001.50c    2.00        $hydrogen
        8016.50c    1.00        $oxygen
mt8     lwtr.01t   $thermal s(a,b) identifier 300K
c
c Samarium-149 (300 K)
m9      62149.50c    1.0
c
c Fission products (300 K)
m10     50120.35c    1.0
c
c Xenon-135 (300 K)
m11     54135.50c    1.0
c
c Xenon-135 (600 K)
m12     54135.53c    1.0
c
c Xenon-135 (900 K)
m13     54135.54c    1.0
c
c Be-9 (300 K)
m14     4009.50c    1.0
c
c Li-6 (300 K)
m15     3006.50c    1.0
c
c U-235 (300 K)
m16     92235.50c    1.0
c
c U-235 (900 K)
m17     92235.54c    1.0
c
c U-238 (300 K)
m18     92238.50c    1.0
c
c U-238 (900 K)
m19     92238.54c    1.0

```

```

c
c Pu-239 (300 K)
m20      94239.55c  1.0
c
c
c TALLIES
c
c fission tallies
f14:n    12 t
f24:n    22 t
f34:n    32 t
f44:n    42 2 t
c
c
c Sm-149(n,g) 300K
f74:n    12 22 32 42 2 t
c
c FP (n,g) 300K
f84:n    12 22 32 42 2 t
c
c Xe-135 (n,g) 300K
f94:n    12 22 32 42 2 t
c
c Xe-135 (n,g) 600K
f104:n   12 22 32 42 2 t
c
c Xe-135 (n,g) 900K
f114:n   12 22 32 42 2 t
c
c Be-9 (n,a) 300K
f124:n   9 t
c
c Be-9 (n,a) 300K alpha prod.
f134:n   9 t
c
c Li-6 (n,T) 300K
f144:n   9 t
c
c Li-6 (n,T) 300K tritium prod.
f154:n   9 t
c
c U-235 (n,a) 300K
f164:n   12 22 32 42 2 t
c
c U-235 (n,a) 900K
f174:n   12 22 32 42 2 t
c
c U-238 (n,g) 300K
f184:n   12 22 32 42 2 t
c
c U-238 (n,g) 900K
f194:n   12 22 32 42 2 t
c
c Pu-239 (n,fp) 300K
f204:n   12 22 32 42 2 t
c
c Pu-239 (n,g) 300K
f214:n   12 22 32 42 2 t
c
c flux
f224:n   12 22 32 42 2 t
c
c
c tally multiplier cards
c
fm14     -1 1 -6
fm24     -1 1 -6
fm34     -1 1 -6
fm44     -1 1 -6
fm74     1.0 9 102      $Sm-149(n,g) 300K
fm84     1.0 10 102     $FP (n,g) 300K

```

fm94	1.0 11 102	\$Xenon-135 (300 K)
fm104	1.0 12 102	\$Xenon-135 (600 K)
fm114	1.0 13 102	\$Xenon-135 (900 K)
fm124	1.0 14 107	\$Be-9 (n,a) 300K
fm134	1.0 14 207	\$Be-9 (n,a) 300K alpha prod.
fm144	1.0 15 105	\$Li-6 (n,T) 300K
fm154	1.0 15 205	\$Li-6 (n,T) 300K tritium prod.
fm164	1.0 16 102	\$U-235 (n,a) 300K
fm174	1.0 17 102	\$U-235 (n,a) 900K
fm184	1.0 18 102	\$U-238 (n,g) 300K
fm194	1.0 19 102	\$U-238 (n,g) 900K
fm204	1.0 20 -6	\$Pu-239 (n,fp) 300K
fm214	1.0 20 102	\$Pu-239 (n,g) 300K
prcnp	50 50 0 1	
print	10 60 100 110	

C.2: MCNP 4A Sample Output File

The contents of the following output file pertain to the input file in Appendix C.1.

```

=====
estimator cycle 410 ave of 400 cycles combination simple average combined average corr
k(collision) 0.991182 1.014158 0.0013 k(col/abs) 1.014335 0.0013 1.014295 0.0013 0.9770
k(absorption) 0.990354 1.014513 0.0013 k(abs/trk ln) 1.013545 0.0011 1.013811 0.0011 0.1564
k(trk length) 0.909672 1.012678 0.0016 k(trk ln/col) 1.013388 0.0011 1.013593 0.0011 0.1685
rem life(col) 5.6250E+04 5.2599E+04 0.0027 k(col/abs/trk ln) 1.013750 0.0011 1.013721 0.0011
rem life(abs) 5.6484E+04 5.2609E+04 0.0027 life(col/abs) 5.2604E+04 0.0027 5.2614E+04 0.0027 0.9987
source points generated 1020

source distribution written to file srcpt cycle = 410
1 problem summary

run terminated when 410 kcode cycles were done.
+ 08/28/98 20:55:02
multiple Cintichem targets, k-effective simulation probid = 08/28/98 17:35:12
0
neutron creation tracks weight energy neutron loss tracks weight energy
(per source particle) (per source particle)

source 410408 9.9901E-01 2.0302E+00 escape 10785 1.0623E-02 2.2341E-03
energy cutoff 0 0. 0.
time cutoff 0 0. 0.
weight window 0 0. 0.
cell importance 0 0. 0.
weight cutoff 0 9.8836E-02 8.9289E-08 weight cutoff 439688 9.8665E-02 7.9770E-08
energy importance 0 0. 0.
dxtran 0 0. 0.
forced collisions 0 0. 0.
exp. transform 0 0. 0.
upscattering 0 0. 6.3714E-07 downscattering 0 0. 1.7153E+00
capture 0 6.6279E-01 1.0227E-01
(n,xn) 80130 1.8078E-01 1.9703E-01 loss to (n,xn) 40065 9.0391E-02 4.0485E-01
fission 0 0. 0. loss to fission 0 4.1625E-01 2.6138E-03
total 490538 1.2786E+00 2.2272E+00 total 490538 1.2786E+00 2.2272E+00

number of neutrons banked 40065 average lifetime, shakes cutoffs
neutron tracks per source particle 1.1952E+00 escape 1.1537E+05 tco 1.0000E+34
neutron collisions per source particle 2.9774E+02 capture 4.7517E+04 eco 0.0000E+00
total neutron collisions 122194568 capture or escape 4.8172E+04 wvc1 -5.0000E-01
net multiplication 1.0905E+00 0.0004 any termination 5.3219E+04 wvc2 -2.5000E-01

computer time so far in this run 199.82 minutes maximum number ever in bank 3
computer time in mcrun 198.91 minutes bank overflows to backup file 0
source particles per minute 2.0632E+03 field length 0
random numbers generated 868040443 most random numbers used was 28863 in history 143492

keff estimator keff standard deviation 68 % confidence 95 % confidence 99 % confidence
collision 1.01416 0.00129 1.01287 to 1.01545 1.01159 to 1.01673 1.01075 to 1.017
absorption 1.01451 0.00130 1.01321 to 1.01581 1.01193 to 1.01710 1.01109 to 1.017
track length 1.01258 0.00165 1.01093 to 1.01422 1.00930 to 1.01586 1.00823 to 1.016
col/absorp 1.01430 0.00129 1.01300 to 1.01559 1.01172 to 1.01687 1.01089 to 1.017
abs/trk len 1.01381 0.00110 1.01271 to 1.01491 1.01163 to 1.01599 1.01092 to 1.016
col/trk len 1.01359 0.00110 1.01249 to 1.01469 1.01141 to 1.01578 1.01070 to 1.016
col/abs/trk len 1.01372 0.00110 1.01262 to 1.01482 1.01164 to 1.01590 1.01083 to 1.016

if the largest of each keff occurred on the next cycle, the keff results and 68, 95, and 99 percent confidence interval

keff estimator keff standard deviation 68 % confidence 95 % confidence 99 % confidence
collision 1.01436 0.00130 1.01306 to 1.01566 1.01176 to 1.01696 1.01092 to 1.017
absorption 1.01469 0.00131 1.01339 to 1.01600 1.01209 to 1.01730 1.01124 to 1.018
track length 1.01282 0.00166 1.01116 to 1.01448 1.00952 to 1.01612 1.00844 to 1.017
col/abs/trk len 1.01395 0.00111 1.01284 to 1.01507 1.01174 to 1.01617 1.01101 to 1.016

the estimated collision/absorption neutron lifetimes, one standard deviations, and 68, 95, and 99 percent confidence in

type lifetime(sec) standard deviation 68 % confidence 95 % confidence 99 % con
removal 5.2614E-04 1.4242E-06 5.2471E-04 to 5.2756E-04 5.2330E-04 to 5.2898E-04 5.2238E-04 t
capture 4.7572E-04 1.1765E-06 4.7454E-04 to 4.7690E-04 4.7338E-04 to 4.7806E-04 4.7261E-04 t
fission 2.3016E-04 7.1807E-07 2.2945E-04 to 2.3088E-04 2.2873E-04 to 2.3160E-04 2.2827E-04 t
escape 1.1622E-03 1.1934E-05 1.1502E-03 to 1.1741E-03 1.1384E-03 to 1.1880E-03 1.1307E-03 t
1 average individual and combined collision/absorption/track-length keff results for 13 different batch sizes

```

1tally 14 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 total
1.22574E+00 1.22574E+00

cell 12
multiplier bin: -1.00000E+00 1 -6
1.10202E-02 0.0092

cell union total
multiplier bin: -1.00000E+00 1 -6
1.10202E-02 0.0092

1tally 24 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 22 total
1.22574E+00 1.22574E+00

cell 22
multiplier bin: -1.00000E+00 1 -6
6.31986E-02 0.0041

cell union total
multiplier bin: -1.00000E+00 1 -6
6.31986E-02 0.0041

1tally 34 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 32 total
1.22574E+00 1.22574E+00

cell 32
multiplier bin: -1.00000E+00 1 -6
1.13376E-01 0.0029

cell union total
multiplier bin: -1.00000E+00 1 -6
1.13376E-01 0.0029

1tally 44 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 42 2 total
1.22574E+00 1.22574E+00 2.45148E+00

cell 42
multiplier bin: -1.00000E+00 1 -6
1.42674E-01 0.0028

cell 2
multiplier bin: -1.00000E+00 1 -6
8.65886E-03 0.0106

cell union total
multiplier bin: -1.00000E+00 1 -6
7.56653E-02 0.0025

1tally 74 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 9 102
2.48714E+01 0.0102

cell 22
multiplier bin: 1.00000E+00 9 102
1.43211E+02 0.0045

cell 32
multiplier bin: 1.00000E+00 9 102
2.56144E+02 0.0032

cell 42
multiplier bin: 1.00000E+00 9 102
3.23368E+02 0.0028

cell 2
multiplier bin: 1.00000E+00 9 102
1.94256E+01 0.0114

cell union total
multiplier bin: 1.00000E+00 9 102
1.53404E+02 0.0018

1 tally 84 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 10 102
3.46527E-02 0.0090

cell 22
multiplier bin: 1.00000E+00 10 102
1.98964E-01 0.0040

cell 32
multiplier bin: 1.00000E+00 10 102
3.56521E-01 0.0028

cell 42
multiplier bin: 1.00000E+00 10 102
4.46574E-01 0.0025

cell 2
multiplier bin: 1.00000E+00 10 102
2.70630E-02 0.0104

cell union total
multiplier bin: 1.00000E+00 10 102
2.12755E-01 0.0016

1 tally 94 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 11 102
9.99889E+02 0.0096

cell 22
multiplier bin: 1.00000E+00 11 102
5.76142E+03 0.0043

cell 32
multiplier bin: 1.00000E+00 11 102
1.03313E+04 0.0030

cell 42
multiplier bin: 1.00000E+00 11 102
1.31159E+04 0.0027

cell 2
multiplier bin: 1.00000E+00 11 102
7.89633E+02 0.0108

cell union total
multiplier bin: 1.00000E+00 11 102
6.19962E+03 0.0017

1 tally 104 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 12 102
1.00088E+03 0.0086

cell 22
multiplier bin: 1.00000E+00 12 102

5.76700E+03 0.0043

cell 32
multiplier bin: 1.00000E+00 12 102
1.03417E+04 0.0030

cell 42
multiplier bin: 1.00000E+00 12 102
1.31295E+04 0.0027

cell 2
multiplier bin: 1.00000E+00 12 102
7.90500E+02 0.0108

cell union total
multiplier bin: 1.00000E+00 12 102
6.20593E+03 0.0017

1 tally 114 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 13 102
1.00182E+03 0.0096

cell 22
multiplier bin: 1.00000E+00 13 102
5.77227E+03 0.0043

cell 32
multiplier bin: 1.00000E+00 13 102
1.03518E+04 0.0030

cell 42
multiplier bin: 1.00000E+00 13 102
1.31425E+04 0.0027

cell 2
multiplier bin: 1.00000E+00 13 102
7.91323E+02 0.0108

cell union total
multiplier bin: 1.00000E+00 13 102
6.21190E+03 0.0017

1 tally 124 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 9 total
5.66464E+05 5.66464E+05

cell 9
multiplier bin: 1.00000E+00 14 107
6.78398E-07 0.0026

cell union total
multiplier bin: 1.00000E+00 14 107
6.78398E-07 0.0026

1 tally 134 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 9 total
5.66464E+05 5.66464E+05

cell 9
multiplier bin: 1.00000E+00 14 207
6.78398E-07 0.0026

cell union total
multiplier bin: 1.00000E+00 14 207
6.78398E-07 0.0026

1 tally 144 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 9 total
5.66464E+05 5.66464E+05

cell 9
multiplier bin: 1.00000E+00 15 105
1.79751E-01 0.0021

cell union total
multiplier bin: 1.00000E+00 15 105
1.79751E-01 0.0021
1 tally 164 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 9 total
5.66464E+05 5.66464E+05

cell 9
multiplier bin: 1.00000E+00 15 205
1.79751E-01 0.0021

cell union total
multiplier bin: 1.00000E+00 15 205
1.79751E-01 0.0021
1 tally 164 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 16 102
3.48989E-02 0.0093

cell 22
multiplier bin: 1.00000E+00 16 102
1.98780E-01 0.0040

cell 32
multiplier bin: 1.00000E+00 16 102
3.56618E-01 0.0028

cell 42
multiplier bin: 1.00000E+00 16 102
4.43854E-01 0.0026

cell 2
multiplier bin: 1.00000E+00 16 102
2.70896E-02 0.0105

cell union total
multiplier bin: 1.00000E+00 16 102
2.12248E-01 0.0016
1 tally 174 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 17 102
3.49115E-02 0.0091

cell 22
multiplier bin: 1.00000E+00 17 102
1.99107E-01 0.0040

cell 32
multiplier bin: 1.00000E+00 17 102
3.57082E-01 0.0028

cell 42
multiplier bin: 1.00000E+00 17 102
4.44211E-01 0.0025

cell 2
multiplier bin: 1.00000E+00 17 102
2.71023E-02 0.0104

cell union total
multiplier bin: 1.00000E+00 17 102
2.12483E-01 0.0016
1 tally 184 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons

number of histories used for normalizing tallies = 400000.00

volumes

cell:	12	22	32	42	2	total
	1.22574E+00	1.22574E+00	1.22574E+00	1.22574E+00	1.22574E+00	6.12870E+00

cell 12

multiplier bin: 1.00000E+00 18 102
9.13118E-03 0.0800

cell 22

multiplier bin: 1.00000E+00 18 102
5.02568E-02 0.0318

cell 32

multiplier bin: 1.00000E+00 18 102
9.33507E-02 0.0264

cell 42

multiplier bin: 1.00000E+00 18 102
9.93145E-02 0.0230

cell 2

multiplier bin: 1.00000E+00 18 102
7.12989E-03 0.1012

cell union total

multiplier bin: 1.00000E+00 18 102
5.18368E-02 0.0149

1 tally 194 nps = 410408

tally type 4 track length estimate of particle flux.
tally for neutrons

number of histories used for normalizing tallies = 400000.00

volumes

cell:	12	22	32	42	2	total
	1.22574E+00	1.22574E+00	1.22574E+00	1.22574E+00	1.22574E+00	6.12870E+00

cell 12

multiplier bin: 1.00000E+00 19 102
9.34438E-03 0.0853

cell 22

multiplier bin: 1.00000E+00 19 102
5.18067E-02 0.0263

cell 32

multiplier bin: 1.00000E+00 19 102
9.53423E-02 0.0226

cell 42

multiplier bin: 1.00000E+00 19 102
1.02153E-01 0.0194

cell 2

multiplier bin: 1.00000E+00 19 102
7.18473E-03 0.0803

cell union total

multiplier bin: 1.00000E+00 19 102
5.31262E-02 0.0126

1 tally 204 nps = 410408

tally type 4 track length estimate of particle flux.
tally for neutrons

number of histories used for normalizing tallies = 400000.00

volumes

cell:	12	22	32	42	2	total
	1.22574E+00	1.22574E+00	1.22574E+00	1.22574E+00	1.22574E+00	6.12870E+00

cell 12

multiplier bin: 1.00000E+00 20 -6
3.15062E-01 0.0087

cell 22

multiplier bin: 1.00000E+00 20 -6
1.80788E+00 0.0043

cell 32

multiplier bin: 1.00000E+00 20 -6
3.22722E+00 0.0031

cell 42

multiplier bin: 1.00000E+00 20 -6
3.99018E+00 0.0027

cell 2

multiplier bin: 1.00000E+00 20 -6
2.44287E-01 0.0123

cell union total
multiplier bin: 1.00000E+00 20 -8
1.91893E+00 0.0017
1 tally 214 nps = 410408
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
multiplier bin: 1.00000E+00 20 102
1.46463E-01 0.0108

cell 22
multiplier bin: 1.00000E+00 20 102
8.40674E-01 0.0048

cell 32
multiplier bin: 1.00000E+00 20 102
1.49500E+00 0.0034

cell 42
multiplier bin: 1.00000E+00 20 102
1.82455E+00 0.0031

cell 2
multiplier bin: 1.00000E+00 20 102
1.12555E-01 0.0148

cell union total
multiplier bin: 1.00000E+00 20 102
8.83847E-01 0.0019

1 tally 224 nps = 410408
tally type 4 track length estimate of particle flux. units 1/cm**2
tally for neutrons
number of histories used for normalizing tallies = 400000.00

volumes
cell: 12 22 32 42 2 total
1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 1.22574E+00 6.12870E+00

cell 12
1.50133E-03 0.0088

cell 22
8.49914E-03 0.0036

cell 32
1.50235E-02 0.0025

cell 42
1.73918E-02 0.0023

cell 2
1.06405E-03 0.0097

cell union total
8.69597E-03 0.0014

.....
dump no. 10 on file runtime nps = 410408 coll = 122194558 ctm = 198.91 nnn = 868040443

59 warning messages so far.

run terminated when 410 kcode cycles were done.

computer time = 199.84 minutes

mcnp version 4a 10/01/93 08/28/98 20:55:03 probid = 08/28/98 17:35:12

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