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Fuel burnup simulation and analysis of the Missouri S&T Reactor

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FUEL BURNUP SIMULATION AND ANALYSIS OF THE MISSOURI S&T REACTOR

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ABSTRACT

The purpose of this work is to simulate the fuel burnup of the Missouri S&T Reactor. This work was accomplished using the Monte Carlo software MCNP. The primary core configurations of MSTR were modeled and the power history was used to determine the input parameters for the burnup simulation. These simulations were run to determine the burnup for each fuel element used in the core of MSTR.

With these simulations, the new predicted isotopic compositions were added into the model. New core configurations were determined, and the burnup corrected model was used to predict the excess reactivity and control rod worth of the three shim rods in the new configurations labeled 125 and 126. The reactor core was arranged to these configurations in order to determine excess reactivity and control rod worth for the shim rods. When core 125 was tested, the reactor could not attain criticality without external source, which was predicted by the simulations. Core 126 had sufficient excess criticality to support measurement. Those measurements had inconsistent results, and indicated methodology errors.

Based on this work, it is recommended to revise the code for temperature gradients and Doppler effects. The experimental methodology for the rod drop tests should also be revised to ensure the methods are applicable to core parameters. These corrections allow the model and the reactor itself to better reflect each other so that the predictions by MCNP will better reflect the measurements at MSTR.

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TABLE OF CONTENTS

Page

LIST OF FIGURES

LIST OF TABLES

1. INTRODUCTION

For many years, the Missouri S&T Reactor (MSTR) has been an important part of the curriculum of the nuclear engineering program. Every nuclear engineering student works with the reactor at least once in the program, and the facility plays an important role in several nuclear engineering courses. This project deals entirely with MSTR, and is motivated by its operational use and practices, with the hope of improving operational use and management of the reactor as an experimental facility for future nuclear engineering students.

1.1. OVERVIEW OF THE MISSOURI S&T REACTOR

The Missouri S&T Reactor is a 200-kilowatt (kW) pool-type research reactor at the Missouri University of Science and Technology [1]. Built in 1961, the reactor is primarily used for research and training, and allows students to have hands-on, in-class experience with a nuclear reactor [1]. The reactor is also available to outside experimenters and tour groups [1]. Figure 1.1. shows the core of MSTR [1].

The reactor core is located in a pool of water that is 19 feet (ft.) in length, has a width of 9 ft., and has a depth between 27 and 30 ft. [1]. At one end of the pool is the fuel storage pit, which is separated from the rest of the pool by a bulkhead [1]. The core sits on an aluminum grid plate that is suspended from a movable bridge [1]. The reactor uses four control rods [1]. Three of these control rods, known as shim rods, consist of borated stainless steel, and are used for safety and coarse power control [1]. These control rods are connected to their drive mechanisms by electromagnets [1]. The electromagnets hold the rods in place for top-insertion into the core. During emergency shut-down, known as

a SCRAM, magnet power is cut off, causing the rods to drop into the core [1]. The fourth control rod, the regulating rod, is a hollow stainless-steel tube, and is used for fine power adjustment [1].

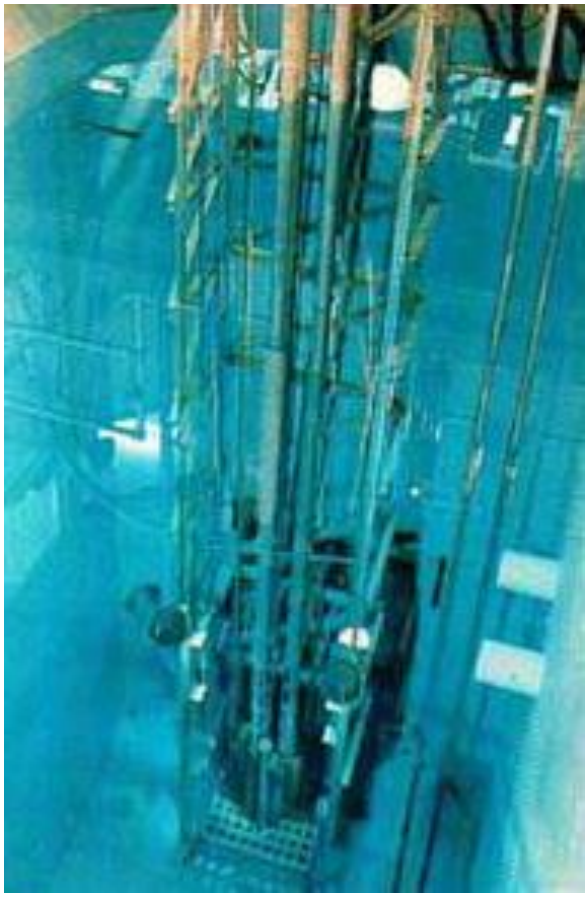

Figure 1.1. The Missouri S&T Reactor [1]

There are several facilities that are available at the reactor for use by students and experimentalists. These include pneumatic transfer tubes (rabbit tubes) that allow samples to be quickly moved in and out of the core, and are slotted into the grid plate [1]. Samples may also be irradiated using the beam port and the thermal column, located at the end of the pool, next to the core [1]. The beam port allows for irradiation of samples

with higher energy neutrons, while the thermal column allows for lower energy neutron irradiation [1]. By moving the bridge, the core can be moved closer to the thermal column to allow for its use, as well as to increase the reflection of neutrons back into the core [1]. When the core is up against the thermal column, the core is said to be in "T" mode, while "W" mode is when the core is moved away from the thermal column [1]. Other experimental facilities or setups may be inserted into core or placed on the grid plate near the fuel [1].

Several instrumentation systems are used to operate and control the reactor. For reactor startup, a fission chamber detector is used to track the log count rate [2]. Two compensated ion chambers (CIC) are also used [2]. One CIC, known as the linear channel, is used to track percent power on a series of set linear scales with the lowest being 2W, and the highest being 200kW [2]. This channel also includes an auto controller that can be set to maintain a stable power by controlling the regulating rod [2]. The second CIC, known as the log and linear channel, tracks percent power of 200kW on a logarithmic scale [2]. This channel also tracks reactor period, the time needed for reactor power to change by a factor of the Euler number, "e," roughly equivalent to 2.71 [2]. Finally, two safety channels, each using an uncompensated ion chamber (UIC), measure percent absolute power [2]. These systems work together to provide safety and redundancy in determining the operating conditions of the nuclear reactor [2].

1.2. FUEL USED BY MSTR

The fuel used by MSTR is in the form of curved parallel plates arranged into a fuel assembly. This type of fuel is known as the Materials Testing Reactor (MTR) type of fuel [2]. When the reactor was built in 1961, the core was originally loaded with highly

enriched uranium (HEU) [2]. In 1992, the fuel was replaced with low enriched uranium (LEU) [2]. The fuel material itself is U_3Si_2-Al, with the uranium enriched to 19.75 percent ^{235}U [2]. The plates are approximately 2 feet (ft) long [2]. The fuel material is then clad in 6061 aluminum alloy A diagram of the fuel plates can be seen in Figure 1.2.

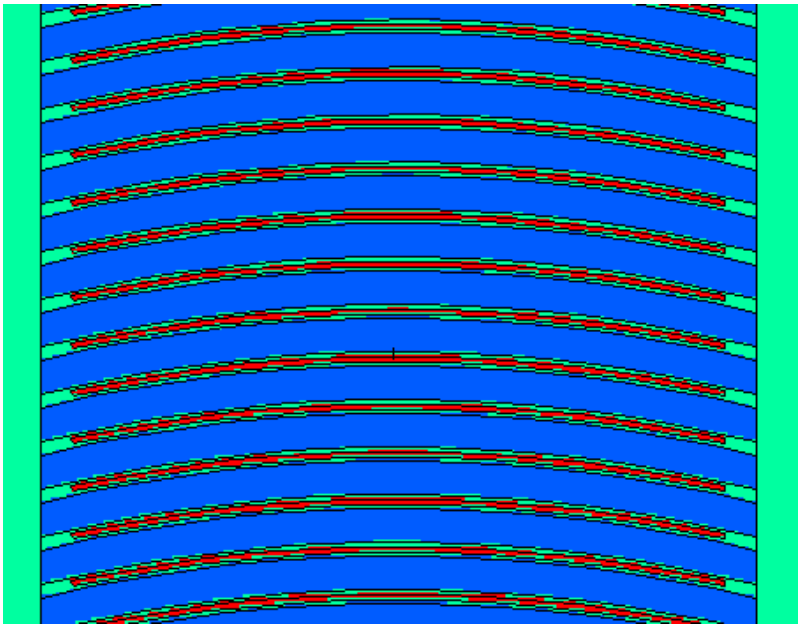

Figure 1.2. MCNP Cross-Section of Fuel Plates [3]

Standard fuel elements consist entirely of fuel plates mounted parallel to each other between a pair of aluminum side plates [2]. A cylindrical nose cone is attached to the bottom of the elements, allowing for the elements to slot into the grid plate [2]. Fuel elements also have a slot located at the top of the elements that allow for fuel handling tools to hook onto the fuel element [2]. This brings the profile of a fuel element to roughly three feet tall, and a roughly square cross-section of three inches on each side [2]. The fuel elements have a defined front and back, with the rear of the element being

concave [2]. The front of the element has the convex side out [2]. Standard fuel elements make up the bulk of the fuel in the core. A diagram of a standard fuel element is given in Figure 1.3.

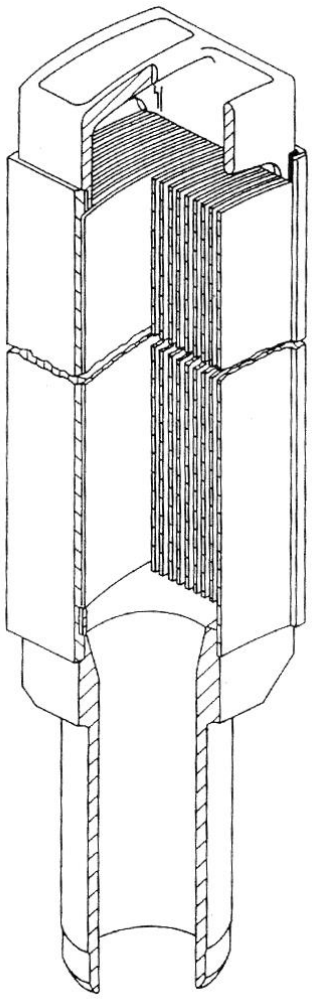

Figure 1.3. Diagram of a Standard Fuel Element [2]

Control rod fuel elements are similar to standard fuel elements, except the central plates are removed to accommodate a rectangular guide tube which the control rods move through [2]. The guide tubes prevent the control rods from contacting the fuel plates during control rod movement and during SCRAMs [2]. The core contains four control

elements, one for the three shim rods, and one for the regulating rod [2]. The reactor also has a fifth, unused, control rod element available, should it be necessary to switch out one of the control elements in the core. A diagram of a control element is given in Figure 1.4.

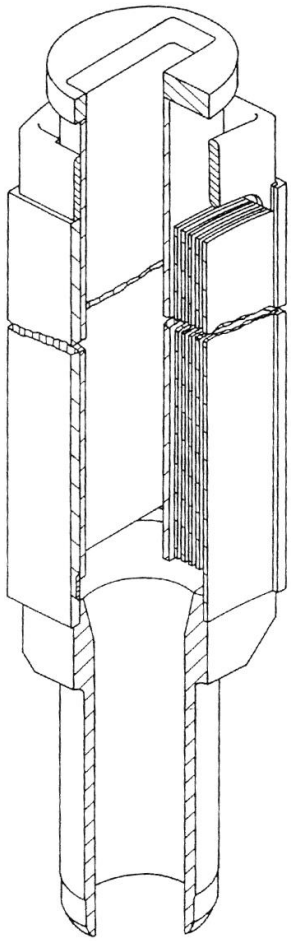

Figure 1.4. Diagram of a Control Rod Fuel Element [2]

Half fuel elements are identical to standard fuel elements, except only half of the plates contain fuel material [2]. The other half are “dummy” plates made of solid aluminum [2]. Either the front plates have fuel, or the back plates have fuel, and are

labeled half front, or half rear elements respectively [2]. The reactor has two half front elements and two half rear elements available for use [2].

The reactor also has an irradiation fuel element available for use [3]. This element has several of the plates removed to form a gap. [3] The plates bordering the gap are replaced with "dummy" plates [2]. This leaves only about half of the plates containing fuel [3]. The removal of these plates allows for a sample to be placed inside the fuel element for irradiation inside the core [2].

1.3. FUEL BURNUP CALCULATION AND PROJECT OBJECTIVE

Over the lifetime of the fuel, the reactor operating staff has tracked the burnup of the fuel [4]. Fuel burnup is the measure of the amount of energy released per unit mass of a given quantity of nuclear fuel, along with tracking the depletion of fissile material and the buildup of fission products [5]. Burnup has the units of megawatt-days (MWd) [5]. Every March, the burnup of the MSTR fuel is calculated by using the power history of the reactor as recorded in the reactor's log books over the course of a year to calculate the amount of fission energy produced during that year [4]. Using that value for energy, the number of grams of ^{235}U consumed is calculated [4].

However, this approach does not account for core layout, spatial distribution of fissile material depletion, actinide creation, and the buildup of fission products [4]. The facility does not track the burnup of individual fuel elements [4]. Instead, the calculated amount of ^{235}U consumed by the entire core is rounded to the nearest gram, deducted from an arbitrary element [4]. As fuel depletion is distributed across the entire core, this method does not accurately indicate amount of fissile material burned or remaining in each fuel element. As several operating parameters of the core are dependent on fuel

burnup, any predictions about how the core may behave may have significant errors. Similar predictions assuming that the fuel will behave the same as if it were fresh will also be off. Therefore, a new method of tracking and calculating fuel burnup is needed.

In 2008, a former member of the Missouri S&T faculty, Dr. Jeffery King, developed a model of MSTR in a nuclear modeling code, MCNP [3]. Previous projects conducted to determine the accuracy of the model compared to physical measurements done on the reactor itself indicate discuss that fuel burnup is a source of error [6], [7]. The model currently simulates conditions in the reactor as if the fuel was still fresh [3]. By 2008, most of the fuel had been in use since 1992 [8]. This means that the fuel burnup may be significant enough to potentially alter the conditions in the reactor core, and cause significant deviation of the model's prediction from physical measurements.

Therefore, the objective of this work is to analyze the power history of MSTR, and use that analysis to simulate the fuel burnup in MCNP. Once the burnup simulation is complete, the core will be rearranged into a new configuration, and MCNP will be used to predict how that core will behave. Those predictions will then be compared to measurements made on the core itself to determine how well the model compares to the measurements. If the model and the measurements agree, the reactor staff could then continue to use MCNP to track fuel burnup in the future, and determine future core configurations from predictions. Otherwise, sources of error in the code must be investigated, along with any issues in the methodology in the core measurements.

2. SIMULATION METHODOLOGY

To understand the methods used for predicting the fuel burnup, an overview of the software MCNP is necessary. The MSTR model was created specifically for MCNP, and therefore plays a key role in predicting the burnup and core parameters of MSTR [3]. The MCNP software has several abilities that play important roles in the model of MSTR.

2.1. MCNP BACKGROUND

Monte Carlo N-Particle transport code (MCNP) is the simulation software used for particle transport, and has the ability to predict the burnup and core parameters of MSTR. MCNP allows for the modeling of complex systems and simulating particle transport through those systems using stochastic method [9]. As the model for this project is about predicting the effect of fuel burnup on the MSTR core parameters, MCNP will be used to model the interactions of neutrons pertaining to criticality and burnup. This will involve the use of criticality subroutine KCODE, and the burnup subroutine CINDER 90 [9].

2.1.1. The KCODE Subroutine. The KCODE subroutine performs iterative calculations in order to calculate the effective neutron multiplication factor, k_{eff}, and requires the KCODE card in the model [9]. The user specifies the number of neutron histories to track [9]. The user then gives an initial guess for the k_{eff}, with a value of one assuming steady state criticality [9]. The user then specifies the number of iterations to run, along with the number of iterations to discard before the iterations converge [9]. The iterations that are not discarded are called active cycles. During each iteration, the KCODE subroutine tracks the particles, and based off their interactions, the subroutine

calculates the k_{eff} [9]. This is repeated for the user specified number of iterations [9]. The result of the calculation is the average of the k_{eff} values from each iteration along with the standard deviation [9]. An example of a KCODE card is in Figure 2.1.

```
kcode 20000 1.0 15 2515
```

Figure 2.1. Example KCODE Card

In this KCODE card, 22,000 particles (histories) are transported per cycle for a total of 2,515 cycles. The first 15 cycles are discarded. The initial k_{eff} guess is 1.0.

2.1.2. CINDER 90: The Burnup Subroutine. The burnup subroutine CINDER 90 is used to simulate the production of fission product isotopes and actinides in fuel [9]. This subroutine is capable of tracking up to up to 3400 isotopes [9]. The neutron interactions of those isotopes are tracked by using cross-sections based on a 63-group energy spectrum calculated from the reactor system [9]. CINDER 90 is also capable of tracking the decay of isotopes. Utilizing CINDER 90 requires the user to specify a burn card in the model, and fill out the associated keywords: TIME, POWER, PFRAC, MAT, MATVOL, BOPT, AND OMIT [9]. The burn card used for core 101 is given in Appendix A.

To use a burn card, the user must specify several operating parameters of the reactor system. The user must specify the length of each time step to be simulated with the units of days using the TIME keyword [9]. The user must then use the POWER keyword to specify what the full operational power of the system is, with units of megawatts (MW) [9]. For each time step, the user must specify the fraction of full power

at which the reactor is operating through the PFRAC keyword [9]. The user must specify the fuel materials to be burned with the MAT card [9]. In this case, each fuel element in the model has its own fuel material, which is assumed to be within each plate of a given element [3]. If a core configuration has, for example, 15 fuel elements, then the fuel material for each element modeled in the core must be included in the burn card [9]. Then, using the MATVOL keyword, the volume of each material must be specified so that MCNP can calculate the amount of material present [9]. The burn options must be specified with the BOPT keyword. These options include a Q-value multiplier, set to one, the output format for the isotope concentrations, and the option to output the isotope concentration after each time step or at the very end [9]. Included in the burn options is the ability to determine which fission products are tracked [9].

The user can choose between three tiers of fission products to track [9]. Tier 1 allows for tracking 12 fission products [9]. Tier 2 allows the tracking of 87 fission products, including all of the fission products in tier 1 and other significant fission products [9]. The third tier tracks 220 isotopes, including all of the isotopes in tiers 1 and 2 [9]. The tracking of more fission products requires more computational time, so to balance run time and accuracy in isotope concentration, tier 2 was selected for this problem.

2.1.3. Disadvantages with Using MCNP. This calculation method does have its disadvantages. There are hundreds of fission products, and many of those undergo interactions and radioactive decay themselves. While CINDER 90 can track 3,400 isotopes, the simulation may predict the production of isotopes for which cross-section data does not exist. In such cases, MCNP will require physics models for the affect

isotopes [9]. In the absence of both cross-section data and models, the isotopes must be omitted for MCNP to perform the burnup [9]. In addition, even if the third, most comprehensive group of fission products was chosen for this simulation, there are still several fission products outside of those [9]. The isotopic composition becomes approximated down to about 100 isotopes [9]. In reality, there could be thousands of isotopes in the fuel. If MCNP predicts that the simulation may model an isotope that MCNP does not have cross-sections for, the simulation will fail [9]. This requires the use of the OMIT keyword, which tells MCNP to ignore specified isotopes [9]. To use the OMIT keyword, the user must specify how many isotopes to omit, what those isotopes are, and from what material the omission must take place [9].

MCNP's calculated isotopic concentrations are averaged over the entire volume of a uniquely identified material. In cases where all fuel materials in a fuel element are identified by the same material number, the isotopic composition is evenly distributed over the regions occupied by the fuel material. In reality, isotopic compositions vary spatially within a fuel element.

Another approximation made is that MCNP does not use a continuous neutron energy spectrum to calculate the interaction rates and concentrations; the burnup calculation is based on 63 energy groups [9]. Information is inherently lost about the interactions of specific energies, and some of the groups may have slightly over predicted or under predicted flux values and concentrations.

2.2. SIMULATION PROCEDURE, AND POTENTIAL SOURCES OF ERROR

In 2008, Dr. Jeffery King, a former professor of Nuclear Engineering, developed the MSTR MCNP model used in this simulation [3]. This model includes the fuel, the

control rods, the pool, the beam port, the thermal column, and any structural components that are believed to have significant impact on particle transport [3]. Each fuel assembly is individually modeled with each assembly having its own fuel material [3]. All of the geometry is already modeled, and the only modifications made are fuel movement required to model each core configuration, and inclusion of appropriate burn cards.

2.2.1. Simulation Procedure. Each relevant core configuration was simulated using the MSTR model. Each core had its own respective burn card corresponding to the fuel elements in that configuration and the appropriate years of operation, requiring power history analysis. The KCODE card used 20,000 particle histories per cycle with 2,500 active cycles. Starting with the first configuration, core 101, the simulation for that core was allowed to run to completion. Once the first simulation was completed, the isotopic composition of each fuel element was provided in MCNP output, listing each isotope by ZAID, and giving the mass and atom fraction of each isotope [9]. Using these new isotopes and their predicted atom fractions, the fuel material cards in the next core configuration were updated to reflect the current burn-up state. This procedure was carried out for the remaining core configurations, giving a final predicted burnup for the final core as of the end of March 2017.

2.2.2. Previous Benchmarking Efforts and Potential Errors in the Model. Previous efforts have been made to benchmark the MSTR model. One such effort was the comparison of normalized axial flux profile values and control rod heights in the model to physical measurements done by Richardson [6]. In this effort, the model over predicted the k_{eff} of the core [6]. The main source of error during that effort was believed to be the fuel burnup [6]. This is now being addressed as the purpose of this current project.

Richardson also noted possible differences in the temperatures of the fuel and the cladding [6]. The model assumes that the fuel and cladding are the same temperature [6]. The temperature also varies axially along the fuel as well. Richardson assumed a two-section temperature gradient that varies linearly based on thermocouples below and above the core [6]. This assumption most likely not the case. This is a potential source of error because interaction cross-sections may change with respect to temperature in a process known as Doppler broadening [5]. Doppler broadening is an effect where the cross-section resonances become shorter and wider, leading to increased resonance absorption and a decreased k_{eff} [5].

Another effort, conducted by O'Bryant, attempted to characterize the temperature profile of the reactor in greater detail with a hot channel analysis [7]. However, that has not been incorporated into the models for cores 118, 121 or 122, as that was only done for cores 101 and 120 [7]. Nor was this work incorporated into the available model [3]. It was decided to continue with assuming equal temperature throughout the core for all core configurations.

O'Bryant also attempted to perform a similar fuel burnup analysis of his own [7]. However, that effort stopped at core 120 [7]. Several years have since passed, and the core has been rearranged to configurations 121 and 122 [8]. In addition, O'Bryant, when calculating the averaged powers, incorporated the shutdown durations of zero power. This led to him approximating a year of power history as a single time step of the reactor being at a constant power without shutdowns [7]. While the overall amount of interactions will average out to be the same as in this current work, having the reactor at a constant but lower average power the whole time may inflate the concentration of short-

lived fission product isotopes. Those would then be carried over into the next power time step, and the next core model. By separating the time at power and the shutdown durations, this current effort allows the shorter-lived fission products to decay away first while longer-lived isotopes will average out to be the same.

This current work does not account for any temperature gradients, and assumes even temperatures throughout the core. Neither does this work build off the previous burnup simulations. The new simulations for this project are done from the very beginning of the fuel lifetime. The simulations done previously over predicted the k_{eff}, with the fuel burnup and temperature gradients being considered the most significant sources of error [6], [7].

3. ANALYSIS OF THE MSTR POWER HISTORY

In order to use CINDER 90 for modeling the burnup of the MSTR fuel, the power history of MSTR must by analyzed. Each core configuration must be analyzed to determine what periods each core was used. The power history must then be compressed to produce usable values for the TIME and PFRAC cards. These time steps are then sorted by core so that the burnup simulation for each core can be run sequentially to model the changes in the MSTR core over time.

3.1. CORE CONFIGURATIONS ANALYZED

Over the course of the reactor's history since acquiring the LEU fuel, the MSTR core has been arranged into 25 configurations, labeled numbers 100 through 124 [8]. Of these configurations, 101, 118, 120, 121, and 122 were used for the longest periods, and were the focus of this analysis [8]. The other core configurations either were used only for a short amount time, such as a few weeks, or were derivatives of more highly used configurations, where the differences were small, such as the swapping positions of two elements. The time period of focus is from July 1992, when the fuel was acquired, to March 2017, when the reactor staff performed its annual burnup calculation for the previous 12 months. All core configurations are shown in Figure 3.1.

3.1.1. Core 101. Core 101was used from July 1992 to August 2009, making this configuration the longest used core configuration with the fuel low enriched fuel to date [8]. Derivatives of this core configuration include core 100, as well as cores 102 through 116, and only differ by one element, and are used for such short amounts of time that the power histories for those configurations can be lumped into the history for core 101 [8].

3.1.2. Core 118. Core 118 was next configuration used for a significant period of time, immediately following core 101. Used from August 2009 to August 2010, this configuration is very similar to core 101, with the main difference being the addition of a half element on the right side of the core [8]. This core configuration was only used for about a year [8]. However, that is long enough for the fuel to undergo significant burnup. Therefore, this core configuration was accounted for in the burnup simulation.

3.1.3. Cores 120 and 121. After a brief test of a core labeled 119, core 120, was the next main core configuration. With this core, the staff shifted the fuel closer towards the beam port [8]. This core was used until January 2014, when fuel element F11 was switched out for fuel element F1, creating core 121 [8]. The reactor had this core until September 2014, where the core was briefly changed back to core 120 for a few weeks before being changed again [4].

3.1.4. Core 122. Core 122 is the final core configuration considered for the burnup simulation. This core was used from October 2014 through March of 2017, which is the endpoint for the burnup simulation. This core was formed from swapping the positions of fuel elements F1 and F11 in core 121 [8]. As this core was used for a few months after March 2017, the burnup simulation does not account for this additional use. However, the simulation does account for the bulk of the time period that core 122 was used.

3.2. POWER HISTORY COMPRESSION

The power history of MSTR is the records containing every duration that the reactor was at a stable power, undergoing a power change, or shutdown. If there were only a few time steps with no power changes, the durations and powers could be used

directly. Because MSTR does not operate at a single power over a single duration, efforts must be taken to compress the power history into usable values that can be plugged into the PFRAC and TIME cards.

101

	1	2	3	4	5	6	7	8	9
A									
B					S				
C				F8	F4	C4			
D			F13	C1	F3	F2	F12	F15	
E			F10	C2	F1	C3	F9	F14	
F			CRT	F5	F6	F7	BRT		

118

	1	2	3	4	5	6	7	8	9
A									
B					S				
C				F8	F4	C4			
D			F13	C1	F3	F2	F12	F15	
E			F10	C2	F1	C3	F9	F14	HF1
F			CRT	F5	F6	F7	BRT		

120

	1	2	3	4	5	6	7	8	9
A									
B						S			
C					C4	F5	F1	F17	
D				F4	F8	F14	C1	F10	F2
E				F9	C3	F12	C2	F7	F3
F				CRT	F15	HCRT	F13	BRT	F6

121

	1	2	3	4	5	6	7	8	9
A									
B						S			
C					C4	F5	F1	F17	
D				F4	F8	F14	C1	F10	F11
E				F9	C3	F12	C2	F7	F3
F				CRT	F15	HCRT	F13	BRT	F6

122

	1	2	3	4	5	6	7	8	9
A									
B						S			
C					C4	F5	F11	F17	
D				F4	F8	F14	C1	F10	F1
E				F9	C3	F12	C2	F7	F3
F				CRT	F15	HCRT	F13	BRT	F6

Legend	
F	Standard Fuel Element
C	Control Rod Fuel Element
HF/HR	Half Front/Half Rear Fuel Element
BRT	Bare Rabbit Tube
CRT	Cadmium Rabbit Tube
HCRT	Hot Cell Rabbit Tube
IF	Irradiation Fuel Element

Figure 3.1. Core Configurations Modeled for Burnup Simulation

3.2.1. Approach and Methodology for Compression. Over the course of an operational day at the reactor, reactor power may change several times, leading to a highly irregular power history [4]. This is in contrast to a power plant reactor, where the

reactor operates continuously at its full power until it requires refueling. At MSTR, once the reactor has been started from zero power, the power of the reactor may be increased, decreased, and held stable several times in a day [4]. There are also days where the reactor does not operate at all.

Over the course of a year, the reactor could undergo hundreds of power changes, and hundreds of periods of stable power. To simulate the burnup of the reactor without any approximations would require calculating the burnup for each power change and each period of steady state operation. This would add up to thousands of iterations of simulation, which would potentially require thousands of hours for the whole simulation to run. For the simulation to finish within a reasonable time, the power history must be compressed such that the simulation has to run as little time steps as possible while minimizing loss in accuracy.

To do this, the energy released from fission is calculated for each time step. If the power history is plotted on a graph, the energy is given by the area under the curve. The energy from each time step is added together, giving to total energy produced while at power for that year. Dividing by the total time the reactor was at power for that year yields the average power the reactor was operated for the total time at power for that given year. This gives a single time step with a single power over a single continuous period. An example of this can be seen in Figure 3.2.

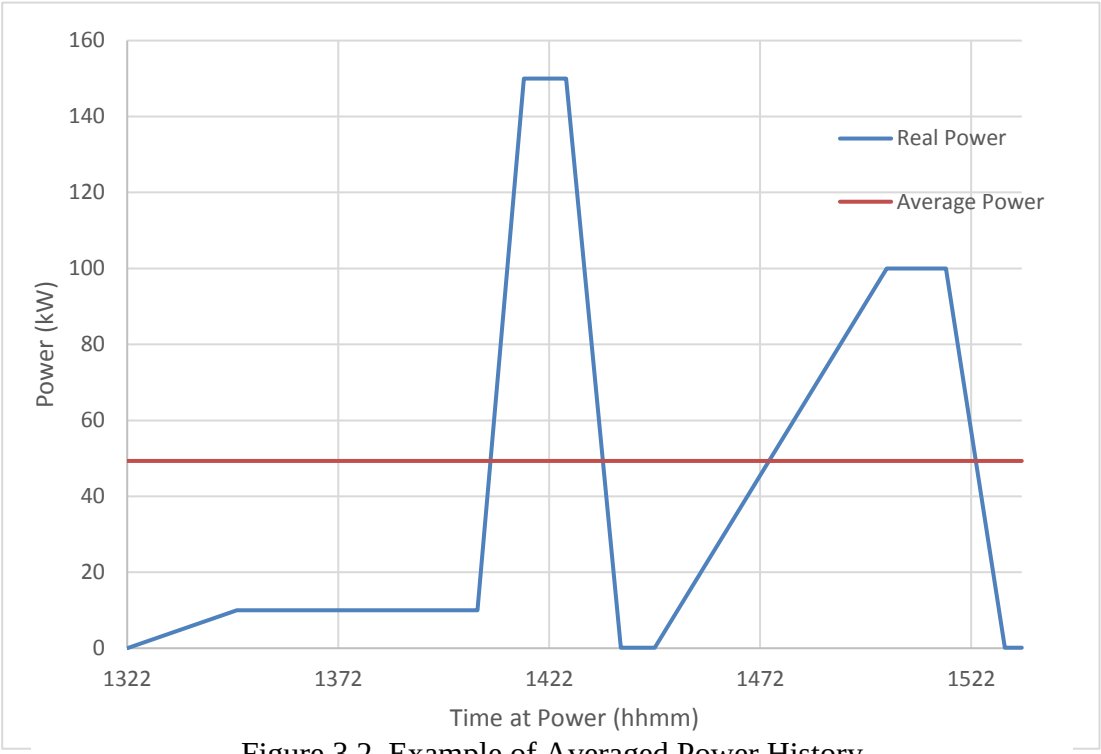

Figure 3.2. Example of Averaged Power History

For the most part, the reactor staff already performs this calculation in its current procedures for calculating fuel depletion and burnup [4]. A slight modification made to them was including the initial startup transient. The calculations previously did not include that initial transient from when the reactor is first started from zero power at the beginning of a day, after a shutdown [4]. These spreadsheets also do not account for core configuration [4]. The staff calculates the fuel depletion at the end of every March. However, the staff may change the core at any point during the year. The core configurations discussed earlier were implemented in the summer and fall months [8]. These periods were calculated as their own time steps and then added to the simulation for the appropriate configuration. This ensures that the entire operation of a core is accounted for in the simulations.

3.2.2. Advantages and Limitations of Power History Compression. This power history provides a key advantage. It was mentioned earlier is condensing the power history into a few time steps would shorten the run time of the simulation. MCNP simulates the state of a reactor system using random numbers to model the movement and interactions of neutrons to determine the criticality of the reactor. The criticality is determined hundreds of times for each time step, converging on a final state of criticality for that time step. However, if this were done for each little power change and steady state operating duration, then this simulation would have to calculate the criticality for thousands of time steps, and would not finish in a timely manner.

Condensing the power down into a few time steps reduces the amount of calculations required in the simulation. This significantly reduces the run time for the simulation. Since the average power comes from the total energy generated from the true power history, the number of fissions is the same. Therefore, both the model and the real reactor should be generating about the same amount of fission products. The rate at which these fission products interact with neutrons should average out to be the same as well. Also, the decay time is the same. Therefore, the isotopic composition at the end of a year should be about the same.

A disadvantage of condensing the power history down to a few time steps is that the power history is heavily approximated. Several power changes and steady state operations are lumped and averaged together to create one power over the entire time at power [4]. Every duration of shutdown is added together to create two time steps of zero power for a whole year. The issue is that lumping the decay time into two steps may lead to the isotopic composition of the fuel in the model to deviate significantly from the real

fuel with the true power history. This simulation operates under the assumption that the isotopic concentrations will not deviate significantly due to compression of the power history. This is a good assumption because the reactor undergoes several days of shutdown, which allows short-live fission products to decay down to insignificance, while the longer-lived isotope concentrations remains relatively unchanged.

4. BURNUP SIMULATION RESULTS AND COMPARISON WITH RECORDS

As mentioned earlier, the methods of fuel burnup calculation do not account for core configuration, resulting in the assignment of the entire yearly depletion into a single random fuel element. As this is essentially a wild guess, the MCNP prediction is expected to deviate significantly from this. The buildup of fission products, which has never been accounted for previously, will also be examined.

4.1. PREDICTED REMAINING MASS OF ^{235}U

A driving factor of this effort is to predict the depletion of fissile material in each fuel element. As such, the burnup and depletion records kept by the MSTR staff will need to be compared to simulation result. This comparison does not account for the production of ^{239}Pu and will be covered later.

4.1.1. Predicted ^{235}U Depletion in MSTR Records. Over the course of the time that the reactor has had the fuel, the staff of MSTR has tracked and calculated the fuel burnup as previously described in Section 1, using the power history spreadsheets analyzed for Section 2. With the rounding of fuel depletion to the nearest gram and assigning that value to a random element in the core, the burnup for each element is hardly accurate. According to the records, some elements have no burnup at all, even though those elements have been used [8]. The burnup of each elements as stated in the MSTR records is given in Table 4.1 [4].

As seen in Table 4.1, fuel elements F6, F17, C3, and C4 have no recorded burnup even though those elements have been in the core for several years. Elements F6, C3, and C4 have been in the core since the new fuel was received [8]. These elements should

have significant burnup by the time of this comparison. Because of this method, the true burnup and isotopic composition is completely unknown. Without a prediction of what the fuel composition of each element is, the staff must resort to trial and error in determining fuel movement and usable future core configurations.

Table 4.1. ^{235}U Depletion as Recorded in MSTR Logs

Fuel Element	Element Use by Core, Noted by an x					Predicted Total U-235 Burned (grams)
	101	118	120	121	122	
F1	x	x	x	x	x	1
F2	x	x	x			1
F3	x	x	x	x	x	1
F4	x	x	x	x	x	1
F5	x	x	x	x	x	1
F6	x	x	x	x	x	
F7	x	x	x	x	x	1
F8	x	x	x	x	x	1
F9	x	x	x	x	x	1
F10	x	x	x	x	x	1
F11				x	x	1
F12	x	x	x	x	x	1
F13	x	x	x	x	x	1
F14	x	x	x	x	x	1
F15	x	x	x	x	x	1
F16						
F17			x	x	x	
F18						
HF1		x				
HF2						
HR1						
HR2						
IF1						
C1	x	x	x	x	x	1
C2	x	x	x	x	x	1
C3	x	x	x	x	x	
C4	x	x	x	x	x	
C5						

4.1.2. MCNP Predicted ^{235}U Depletion. The burnup simulations were performed with the appropriate burn cards for each core. The final isotopic compositions were analyzed, and the ^{235}U depletion in each fuel element was determined, which is given in Table 4.2.

Table 4.2. Final ^{235}U Depletion Results from MCNP

Fuel Element	Fuel Material	Mass of U-235 Burned (grams)	Burnup (GWd/MTHM)
F1	1300	1.8	1.0865
F2	1500	1.4	0.9105
F3	1700	1.1	0.9927
F4	1900	1.0	0.8382
F5	2100	1.6	0.8264
F6	2300	1.0	0.7787
F7	2500	1.7	0.9039
F8	2700	1.0	0.8413
F9	2900	1.0	0.8089
F10	3100	1.7	0.7848
F11	3300	0.3	0.0776
F12	3500	1.7	0.9606
F13	3700	1.1	0.7820
F14	3900	1.1	0.7487
F15	4100	0.4	0.6189
F16	4300	0.0	0.0000
F17	4500	0.4	0.2619
F18	4700	0.0	0.0000
HF1	5100	0.0	0.0180
HF2	5300	0.0	0.0000
HR1	5500	0.0	0.0000
HR2	5700	0.0	0.0000
IF1	5900	0.0	0.0000
C1	6100	0.6	1.2512
C2	6300	1.0	1.2119
C3	6500	1.0	1.2293
C4	6700	1.0	1.0887
C5	6900	0.0	0.0000

Note that in these results, almost every element used in the core for a significant amount of time has undergone noticeable fuel depletion. The only fuel element that MCNP did not show any depletion in was HF1. This could be potentially due to the fact that HF1 was only used in core 118, which was used for just under a year, and that the element was located on the edge of the core, far from the rest of the fuel.

For the rest of the elements, the depletion ranged from 0.3 to 1.8 grams of ^{235}U burned. The most burned element was F1, at 1.0865 gigawatt days per metric ton of heavy metal (GWd/MTHM), with an MCNP prediction of 1.8 grams burned. It should be noted that the precision on the simulation results is only at 0.1 grams. At the range of powers that MSTR operates, the consumption is low. The MCNP simulation shows that it took about 25 years for F1 to exceed a gigawatt day.

4.2. NOTABLE ISOTOPES

Over the course of the simulations, MCNP has modeled the buildup of several isotopes in each burned fuel element. Uranium can also under go neutron capture as well as fission and other reactions. As fission products build up, those isotopes can also undergo neutron capture and other reactions. Because of this, MCNP now predicts that the composition includes several isotopes beyond the original isotopes that made up the U_3Si_2-Al fuel material.

4.2.1. Predicted Actinides. While ^{235}U will mostly undergo fission, it is possible for that isotope to undergo other reactions such as radiative capture. In addition, ^{238}U will only fission at fast energies, meaning that this isotope will undergo significant radiative capture in a thermal reactor such as MSTR [5]. Because of this, several other actinide isotopes will build up in the fuel, and may have an impact on the performance of the core.

These actinides may also undergo capture or fission if they have the cross-sections to do so. For example, Table 4.3. shows the final actinide concentrations for fuel element F1 after all burnup simulations.

One notable isotope is ^{239}Pu, and may significantly affect the performance of this element. As ^{239}Pu is a fissile isotope, an increase in this isotope will increase the reactivity worth of this element, and the buildup of ^{239}Pu in the core will impact core reactivity parameters such as the delayed neutron fraction β_{eff}. As noted before, though, these results may not necessarily reflect reality, and the error is unknown.

Table 4.3. MCNP Predicted Actinide Concentrations in Fuel Element F1

Isotope	Mass (grams)	Atom Fraction
Pa-231	1.47E-06	8.10E-10
U-234	5.53E-05	3.06E-08
U-236	2.51E-01	1.39E-04
U-238	910.7285	5.03E-01
Np-237	3.25E-04	1.80E-07
Pu-239	1.24E-01	6.82E-05
Pu-240	2.90E-04	1.60E-07

4.2.2. Predicted Non-Actinide Isotopes. When a ^{235}U atom undergoes nuclear fission, the nucleons are typically distributed asymmetrically, with one product atom having more nucleons than the other [5]. A typical fission product distribution will have two peaks, with the lower mass peak having higher yields with atomic masses near the nineties, and the higher mass peak having masses roughly in the 140s [5]. Using fuel element F1 as an example again, a predicted non-actinide isotope distribution after all simulations is given in Figure 4.1.

As seen in Figure 4.1., the predicted fission product distribution for F1 roughly corresponds to a standard fission yield distribution. The higher mass peak is centered around Barium and Cesium, with isotopes of lanthanides such as Neodymium and Samarium also being significant [10]. The lower peak indicates significant quantities of Krypton, Zirconium, and Molybdenum isotopes, along with ^{99}Tc [10]. Seeing this simulation result shows that the predicted distribution matches the shape of the theoretical distribution. However, this simulation result cannot be confirmed, as presently, MSTR has no method of performing a non-destructive fuel assay.

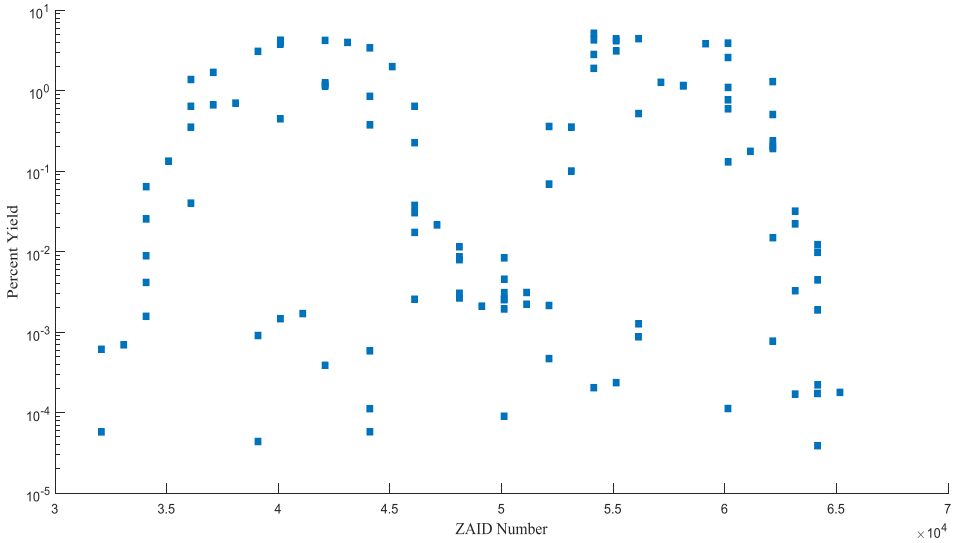

Figure 4.1. Predicted Non-Actinide Distribution in Fuel Element F1

4.3. POTENTIAL SOURCES OF ERROR IN BURNUP RESULTS

Given that the MSTR staff has no available method of measuring the isotopic composition of the fuel material, the true concentrations of actinides and non-actinides in the fuel remains unknown. Since only a few fission products are tracked and several

isotopes are omitted from the simulation, the true composition may include many more isotopes than those that the output predicts.

As the simulations are run, the materials in each plate are assumed to be the same. So, when the simulation is run, MCNP has no way of modeling the spatial distribution between plates in a single element. For example, the rear plates could be more burned than the front ones, or vice versa. The same goes for the isotopic distribution along the length and width of each plate. As the output composition for each simulated core is carried over to the next simulation, this error may add up over time.

5. CORE CHARACTERIZATION METHODOLOGY AND RESULTS

As the fuel composition cannot be directly measured, the performance of the fuel will have to be compared instead. This involves creating new core configurations and simulating the behavior of these new configurations in MCNP. These configurations will then be compared to physical measurements performed on these configurations in the reactor core itself. This involves reactivity measurements, which are used to characterize the reactor core. Such measurements will be the primary parameters for comparison between simulation predictions and experimental measurements.

5.1. REACTOR KINETICS AND THE ROD DROP METHOD

In previous sections, the effective neutron multiplication factor, k_{eff}, was discussed as the parameter that MCNP predicts and tracks over the course of a criticality and burnup simulation. From this parameter, the reactivity can be derived. Any change to a reactor system, such as the insertion of control rods or the depletion of fuel, can be a source of reactivity, so it is important to understand what reactivity is, how reactors are affected by it, and how to measure the reactivity due to inserting control rods or adding fuel.

5.1.1. Reactivity and Reactor Kinetics. Reactivity is defined in terms of k_{eff}. The k_{eff} of a reactor is the ratio of the number of neutrons in the current generation to the number of neutrons in the previous generation [5]. If both values are the same, the k_{eff} is equal to one, and the reactor is critical. A supercritical reactor has a k_{eff} greater than one, and a subcritical reactor has a k_{eff} less than one [5]. Reactivity is a measure of how a change in the reactor system changes the k_{eff}, and is calculated with Equation 1 [5], [11].

$$\rho = \frac{k - k_0}{k} \quad (1)$$

This equation is often simplified by assuming a critical system, where k_{eff} is equal to one, giving Equation 2 [5], [11].

$$\rho = \frac{k - 1}{k} \quad (2)$$

Reactivity can be positive or negative, depending on which direction the k_{eff} changes. Any change in the reactor core may potentially change the k_{eff} of a reactor. Adding, fuel elements, depleting fissile material, and the buildup of fission product poisons will have measurable reactivity effects. Inserting or removing control rods will also have reactivity effects. Changes in temperature, and introducing or removing a void can also add reactivity to a system

The amount of reactivity that a control rod has on the reactor is known as the control rod worth. Knowing the control rod worth allows operators understand how much negative reactivity can be inserted into a reactor. For the core of MSTR, the reactivity worth of each control rod must be measured when changing to a new core configuration [2].

Another important reactivity parameter is the excess reactivity. Excess reactivity is the amount of reactivity in a core above the level that provides exact criticality to the core. If a core without control rods only had enough fuel to reach criticality at its absolute minimum power, that core would have an excess reactivity of zero. Because of this, a core must have positive excess reactivity in order to increase in power. How much excess reactivity a core has can determine how long a core could be used before depletion and

burnup lowers the excess reactivity to zero. For MSTR, the excess reactivity must be calculated along with the control rod worth and is also necessary for characterizing the core [2].

The effects of a reactivity insertion into a reactor core are mathematically described with the point reactor kinetics equations [11]. These equations show how the power of a reactor changes in response to a reactivity insertion, such as a control rod being dropped into the core [11].

$$\frac{dn}{dt} = \left[\frac{\rho(t) - \beta}{\Lambda}\right] n(t) + \sum_{i=1}^{6} \lambda_i C_i(t) \tag{3}$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i \tag{4}$$

Using these equations requires knowing other properties of the core, such as the delayed neutron fraction, the concentration of delayed neutron precursors, and the neutron generation time [11]. If those quantities are known, reactor operators can measure the reactivity inserted by dropping a control rod into the core by measuring reactor power before the drop, and at time intervals afterwards [12].

5.1.2. MSTR Rod Drop Method. The staff of MSTR has a method for measuring control rod worth by rod drops written in the standard operating procedure section 109 [13]. For this procedure, the initial power is 600 Watts [13]. When the rod to be measured is dropped, reactor power is measured 14 seconds and 100 seconds afterwards [13]. Once these measurements are taken, the results are fitted to curves computed by Ohio State

University on a similar reactor system [14]. These curves have power versus reactivity for curves at 14 seconds and 100 seconds [14]. Fitting the measured power to those curves gives the reactivity worth of the measured control rod. As these curves were computed for a different system that may have significantly different core conditions, the applicability of these curves is questionable. One such difference is that these curves were generated at a power of 10 Watts, while the measurements at MSTR are performed at 600 Watts [13], [14]. This higher power may cause Doppler effects to cause MSTR measurements to fall beyond the applicable conditions for using these curves.

These curves are also used for the measurement of excess reactivity, detailed in section 111 of the standard operating procedure [15]. This procedure involves a similar procedure to the rod drop procedure in section 109, except the reactor is made critical at 600 Watts using only one rod, with all other rods fully withdrawn. The excess reactivity is equal to the reactivity worth of the inserted part of the rod. Dropping the rod from this position will give the worth of the rod that was outside the core. Subtracting this value from the total rod worth gives the excess reactivity, as shown in Equation 5 [15].

$$\rho_{ex} = \rho_{total} - \rho_{critical} \tag{5}$$

5.2. CORE BENCHMARKING PROCEDURES AND RESULTS

With the relevant parameters discussed, the benchmarking measurements must be made. New core configurations have to be determined, and then simulated in MCNP to predict the reactivity values for each core. The MSTR core itself must then be rearranged

and control rod worth measurements must then be performed in order to compare these measurements to the predictions given by MCNP for each rod in each core configuration.

5.2.1. Core Configurations Tested. For this comparison, two new core configurations, labeled 125 and 126, were determined based on the burnup results for each element. These configurations are shown in Figure 5.1. The elements that were chosen were predicted to have the least burnup of the used fuel elements. Also included in these configurations were all of the half elements, any fresh full elements, and fresh control element C5. Changes to core layout geometry were made, including modifying control rod placement. Overall geometric symmetry of the layout was also improved.

This core configuration was modified by adding the irradiation fuel element IF1 in between the two half rear elements at the front of the core. This core configuration was labeled core 126. These configurations were designed to minimize neutron leakage from the core, as well as lessen the effects of control rod shadowing. The freshest elements were used in order to lengthen the lifetime of the core configurations, and were spread out to flatten the neutron flux profile of the core.

5.2.2. Core Simulations. Both core configurations for cores 125 and 126 were modeled in the MSTR MCNP model. In order to predict the reactivity worth for each rod, the simulation was run twice for each control rod. First the simulation was run with all control rods fully removed, giving an initial k_{eff}. Then the simulation is then run with the rod to be measured fully inserted, giving a final k_{eff}. These values will then use Equation 1 to compute the predicted reactivity worth of each rod in both experimental core configurations.

125

	1	2	3	4	5	6	7	8	9
A					S				
B					HR1		HR2		
C				F4	C4	F17	C1	F5	
D				F16	F13	F11	F14	F18	
E				F8	C5	F15	C2	F9	
F				CRT	HF1	HCRT	HF2	BRT	

126

	1	2	3	4	5	6	7	8	9
A					S				
B					HR1	IF1	HR2		
C				F4	C4	F17	C1	F5	
D				F16	F13	F11	F14	F18	
E				F8	C5	F15	C2	F9	
F				CRT	HF1	HCRT	HF2	BRT	

Legend	
F	Standard Fuel Element
C	Control Rod Fuel Element
HF/HR	Half Front/Half Rear Fuel Element
BRT	Bare Rabbit Tube
CRT	Cadmium Rabbit Tube
HCRT	Hot Cell Rabbit Tube
IF	Irradiation Fuel Element

Figure 5.1. Experimental Core Configurations

$$\sigma_x = \sqrt{\frac{\sum_{i=1}^{N}(x_i - \overline{x})}{N}} \tag{6}$$

As these are Monte Carlo simulations, it is important make note of the standard deviation for the predictions. For this work, the standard deviation is computed using Equation 6 [16]. Since a standard deviation is given for the k_{eff} values, the reactivity values will have their own standard deviation, propagated from the k_{eff} standard deviations using Equation 7 [16].

$$\sigma_f = \sqrt{(\frac{\partial f}{\partial x})^2\sigma_x^2 + (\frac{\partial f}{\partial y})^2\sigma_y^2 + (\frac{\partial f}{\partial z})^2\sigma_z^2} \tag{7}$$

For the excess reactivity, the k_{eff} values from the simulations of all rods fully withdrawn can be used directly, as they represent the reactivity of a core with no control material. The results of the simulations for each core are given in Table 5.1 and Table 5.2.

Table 5.1. Predicted Excess Reactivity and Rod Worth for Core 125

	ρ (%Δk/k)	std. dev.
Excess Reactivity	-0.22149	0.00011
Worth of Rod 1	-2.52040	0.00016
Worth of Rod 2	-2.47302	0.00016
Worth of Rod 3	-2.44356	0.00016

Table 5.2. Predicted Excess Reactivity and Rod Worth for Core 126

	ρ (%Δk/k)	std. dev.
Excess Reactivity	1.02146	0.00011
Worth of Rod 1	-2.66123	0.00034
Worth of Rod 2	-2.23738	0.00035
Worth of Rod 3	-2.24876	0.00037

Notice that, as shown in Table 5.1., the excess reactivity is negative. This indicates that MCNP is predicting that the core is a subcritical system. Therefore, it would be expected that Core 125 could not reach a critical state without more reactivity such as the startup neutron source inserted into the core.

For the rod worth predictions for core 126, the simulation was only run for 330 cycles. This was done in order to shorten the run time. Given the standard deviations for those runs, the benefit of letting the simulation run for 2500 cycles is insignificant. For future simulations, less active cycles may be used.

A potential problem with this simulation procedure is that MCNP does not simulate a critical state before inserting a rod. This is an imperfection in the comparison, because in the simulation, the effects of the other rods are minimized, where as in the physical measurement, other rods are in the core to bring the reactor to steady state. Because of this, shadowing effects may impact the measured reactivity of the core.

5.2.3. Core Measurement Experiments. The control rod worth of the rods for cores 125 and 126 must then be measured. This involves using the fuel movement procedures found in section 207 of the standard operating procedure [8]. Once core 125 was arranged, the reactor staff attempted to bring the reactor to power. In their attempt to do so, it was discovered that this core configuration needed every control rod including

the regulating rod to be fully withdrawn from the core. Because of this, the reactor could not reach 600 Watts to perform rod drops. As the startup neutron source was in the core during this, it is possible that without it, the reactor could not be brought to power at all. This behavior, however, does line up with the negative excess reactivity predicted by MCNP for this core.

Core 126 was then formed by adding IF1 to the front of the core, providing enough reactivity to the core to attain requisite power levels. Using a procedure based on sections 109 and 111 of the standard operating procedure, a series of rod drops were performed. The reactor was brought to 600 Watts, and then the rod to be measured was then dropped from the fully withdrawn position. Each of the three shim rods were dropped five times in order to check consistency. Once the full drops were completed, each rod was then dropped partially inserted with all others removed, following the excess reactivity procedure. Again, each rod was measured five times to check consistency. The results for each rod are given in Table 5.3.

Table 5.3. Core 126 Rod Drop Measurement Results

	Rod 1 Values (%Δk/k)		Rod 2 Values (%Δk/k)		Rod 3 Values (%Δk/k)	
Drop	Total Worth	Partial Worth	Total Worth	Partial Worth	Total Worth	Partial Worth
1	2.8441	1.2468	2.2338	0.9547	2.1881	0.9685
2	2.7766	1.2530	2.2957	0.9873	2.1974	0.9679
3	2.8304	1.2474	2.3107	0.9826	2.1795	0.9751
4	2.7921	1.2473	2.2908	0.9675	2.2121	0.9629
5	2.8213	1.2503	2.2823	0.9659	2.1657	0.9833
average	2.8129	1.2489	2.2827	0.9716	2.1886	0.9715
std. dev	0.0249	0.0024	0.0261	0.0119	0.0157	0.0070
ρex	1.5640		1.3111		1.2170	
ρ std. dev.	0.0250		0.0287		0.0172	

As seen in Table 5.3., the value of the excess reactivity varies with the rod used in its determination. This is unexpected because excess reactivity is a total core parameter and should have no dependence on which rod was used to measure it. It is unknown why the excess reactivity value varies from rod to rod, but it is believed that there are potential shadowing effects due to the proximity of the rods to each other.

5.3. POTENTIAL SOURCES OF ERROR AND EFFECTS ON RESULTS

Temperature and Doppler effects have already been discussed as possible simulation errors. However, because of the inconsistencies in the measured. excess reactivity values, there must also be a flaw in the methodology for the rod worth measurement. Because of this the results from both the simulation and the measurements are inconclusive.

Another major issue arose in the model with the control rod definitions. It was discovered that the definitions of the control rods in the MSTR model are flawed. This was discovered when the control rods in the model were viewed in the visual editor, and the bottom of the control rods were found to be too high by about 1.15 inches. This is an error that was not discovered before the simulations were performed, and any future work using the MSTR model in MCNP will have to either correct for or account for this. Neither is it known how this error was made. This error may possibly at least partially account for the k_{eff} values predicted in previous works.

5.3.1. Possible Simulation Error Sources. For the simulation, the results could be affected by several factors. The two cores designed for the experiments were chosen to use as much fresh fuel as possible in order to minimize the effects of poison buildup as much as possible. The burned elements used were in the simulations and experiments

were the elements indicated by MCNP to have the least burnup. This could allow for the burnup effects to be accounted for while allowing the effects of other possible errors like temperature and Doppler effects to be visible. The previously mentioned control rod errors may also have an effect on the burnup by affecting fluxes near certain elements.

5.3.2. Potential Experimental Sources of Error. The measurement methodology may also have significant flaws in it. The rod worth curves and the potential Doppler effects have already been discussed in this regard. However, there were also concerns with shadowing effects, where the proximity of control rods to each other affects the ability of them to depress neutron flux. This could potentially affect the reactivity worth of the measured rod, or even potentially block the detectors in the core from the effects of the rod. Efforts should be made to determine the significance of such effects.

6. CONCLUSION

Based on the results, it is difficult to draw any solid conclusions about the effects of the burnup simulation on the MCNP code. The ability of the code to determine suitable core configurations based on excess reactivity and control rod worth also remains inconclusive. Since the MSTR staff has no way of measuring the isotopic composition of the fuel, there is no way to properly benchmark the burnup predictions themselves. However, recommendations can be made to address the possible errors and make corrections to the MSTR burnup records, the MCNP models, and the methodology for measuring control rod worth and excess reactivity.

6.1. SUMMARY OF RESULTS AND POTENTIAL ERROR SOURCES

Using the power history spreadsheets, significantly used core configurations were determined, and modeled using the MSTR MCNP code. These simulations were run showing that fuel element F1 was the most burned element, having burned 1.8 grams of ^{235}U. The simulations also indicate that MSTR has also burned 21.9 grams of ^{235}U over the lifetime of the fuel. Potential errors in the burnup simulation include the tracking of only a few fission products, and the lack of modeling the spatial distribution of burnup between plates in an element.

For the core prediction simulations, the results could be affected by the burnup simulation errors and approximations above. However, the results are impacted by other errors. The effects of temperature gradients and Doppler broadening must also be determined. The fact that these effects were not in the MSTR model may explain why the code was over predicting k_{eff}. The model is written to simulate a low power scenario at

room temperature. At 600 Watts, the power at which reactivity measurements are made, core conditions may deviate from that significantly due to the aforementioned temperature and Doppler effects.

When core 125 was created, the reactor was only able to reach power after fully withdrawing all rods, and could not reach a high enough power for measurement, as predicted in the simulations. Core 126 was measured, and gave three different values for excess reactivity. For the experimental methodology, issues arise with the use of the Ohio State University curves. The core conditions may differ significantly from the conditions in the core of MSTR. The differences in the initial power have already been noted may contribute greatly to this due to the previously mentioned temperature and Doppler effects. Because the delayed neutron fraction and precursor concentrations are unknown, it is difficult to directly determine control rod worth from reactor kinetics as well. The effects of control rod shadowing must also be considered.

6.2. RECOMMENDATIONS FOR FUTURE WORK

Given the importance of tracking fuel burnup, it is important to continue the burnup simulations into the future. The reactor staff must continue to account for startup transients in the power history, and future configurations must be modeled in the MSTR model. It would be recommended to account for the tier three fission products listed in the MCNP manual in the burn card. To determine the accuracy of the burnup calculation, it is recommended that the MSTR staff find a way to perform a fuel assay in order to determine the true fuel composition.

For predictions of excess reactivity and control rod worth, the simulation must also account for Doppler effects and temperature gradients, as these will affect the

predicted reactivity values. Ensuring the correctness of the control rods is also important. The control rod length and placement must be checked for accuracy because of the possibility of having too much or too little control material in the simulated core.

The rod drop procedure must be revised. Measurements of the delayed neutron precursors and delayed neutron fraction must be made, so that more applicable and up-to-date curves may be generated to determine the reactivity inserted by the MSTR control rods. It is recommended that for the rod drop procedure, the initial power should be 10 Watts to minimize temperature and Doppler effects, and that power should be measured in 10 second intervals in order to plot the curve of decreasing reactor power due to the negative reactivity insertion. Rod shadowing effects must be quantified and minimized for the partial drops. This can be done by using control rods that are further away from each other.

Finally, the staff of MSTR should consider the possibility of requesting new fuel. As several of the elements are fresh, not all of the fuel needs to be replaced. However, as more of the fuel becomes poisoned by fission products, the reactivity inserted by them will continue to decrease, and the operators will have issues of maintaining reactor power if the excess reactivity falls too low. If all of the fuel is replaced, there will be a condition for a fresh start on tracking fuel burnup and managing the core with new procedures in place.

APPENDIX A.
EXAMPLE BURN CARD

```
c Burn Card: valid for core 101W and derivatives where core configuration was
c   changed temporarily or where no new elements were added or exchanged.
c
burn  time   = 12.52638889   57.47361111   180    $  July 1992-March 1993
               18.49027778   166.5097222   180    $ April 1993-March 1994
               17.08055556   167.9194444   180    $ April 1994-March 1995
               22.52291667   163.4770833   180    $ April 1995-March 1996
               13.42777778   171.5722222   180    $ April 1996-March 1997
               16.54583333   168.4541667   180    $ April 1997-March 1998
               15.95555556   169.0444444   180    $ April 1998-March 1999
               16.86527778   169.1347222   180    $ April 1999-March 2000
               17.63819444   167.3618056   180    $ April 2000-March 2001
               27.30069444   157.6993056   180    $ April 2001-March 2002
               25.83958333   159.1604167   180    $ April 2002-March 2003
               39.34861111   146.6513889   180    $ April 2003-March 2004
               42.25         142.75        180    $ April 2004-March 2005
               34.08333333   150.9166667   180    $ April 2005-March 2006
               41.46805556   143.5319444   180    $ April 2006-March 2007
               35.98958333   150.0104167   180    $ April 2007-March 2008
               27.88333333   157.1166667   180    $ April 2008-March 2009
               9.106944444   18.89305556   100    $ April 2009-August2009
c
      pfrac  = 0.070856346   0             0      $  July 1992-March 1993
               0.118548544   0             0      $ April 1993-March 1994
               0.076177679   0             0      $ April 1994-March 1995
               0.104902637   0             0      $ April 1995-March 1996
               0.103504454   0             0      $ April 1996-March 1997
               0.101460221   0             0      $ April 1997-March 1998
               0.168458441   0             0      $ April 1998-March 1999
               0.107029899   0             0      $ April 1999-March 2000
               0.088971581   0             0      $ April 2000-March 2001
               0.098852888   0             0      $ April 2001-March 2002
               0.141121187   0             0      $ April 2002-March 2003
               0.20380884    0             0      $ April 2003-March 2004
               0.158252677   0             0      $ April 2004-March 2005
               0.139883753   0             0      $ April 2005-March 2006
               0.16319046    0             0      $ April 2006-March 2007
               0.090933227   0             0      $ April 2007-March 2008
               0.085159941   0             0      $ April 2008-March 2009
               0.160225494   0             0      $ April 2009-August2009
c
      power  = 0.2
```

```
c
      mat    = 1300 1500 1700 1900 2100 2300 2500 2700 2900 3100 3500
               3700 3900 4100 6100 6300 6500 6700
c
      omit   = 1300  6  12027 13026 13028 14027 14031 16031
               1500  6  12027 13026 13028 14027 14031 16031
               1700  6  12027 13026 13028 14027 14031 16031
               1900  6  12027 13026 13028 14027 14031 16031
               2100  6  12027 13026 13028 14027 14031 16031
               2300  6  12027 13026 13028 14027 14031 16031
               2500  6  12027 13026 13028 14027 14031 16031
               2700  6  12027 13026 13028 14027 14031 16031
               2900  6  12027 13026 13028 14027 14031 16031
               3100  6  12027 13026 13028 14027 14031 16031
               3500  6  12027 13026 13028 14027 14031 16031
               3700  6  12027 13026 13028 14027 14031 16031
               3900  6  12027 13026 13028 14027 14031 16031
               4100  6  12027 13026 13028 14027 14031 16031
               6100  6  12027 13026 13028 14027 14031 16031
               6300  6  12027 13026 13028 14027 14031 16031
               6500  6  12027 13026 13028 14027 14031 16031
               6700  6  12027 13026 13028 14027 14031 16031
c
      bopt   = 1 14 0
```

APPENDIX B.
TIER 2 FISSION PRODUCTS

Table B.1. MCNP Tier 2 Fission Products [9]

^{74}As ^{75}As
^{79}Br ^{81}Br
^{78}Kr ^{80}Kr ^{82}Kr ^{83}Kr ^{84}Kr ^{86}Kr
^{85}Rb
^{88}Y ^{89}Y
^{90}Zr ^{91}Zr ^{92}Zr ^{93}Zr ^{94}Zr ^{96}Zr
^{93}Nb
^{95}Mo
^{99}Tc
^{101}Ru ^{103}Ru ^{105}Ru
^{102}Pd ^{104}Pd ^{105}Pd ^{106}Pd ^{108}Pd ^{110}Pd
^{106}Ag ^{109}Ag
^{106}Cd ^{108}Cd ^{110}Cd ^{111}Cd ^{112}Cd ^{113}Cd
^{120}Sn
^{127}I ^{129}I ^{135}I
^{124}Xe ^{126}Xe ^{128}Xe ^{129}Xe ^{130}Xe ^{131}Xe ^{132}Xe ^{134}Xe ^{135}Xe ^{136}Xe
^{133}Cs ^{134}Cs ^{135}Cs ^{136}Cs ^{137}Cs
^{138}Ba
^{141}Pr
^{143}Nd ^{145}Nd ^{147}Nd ^{148}Nd ^{150}Nd
^{147}Pm ^{149}Pm
^{147}Sm ^{149}Sm ^{150}Sm ^{151}Sm ^{152}Sm
^{151}Eu ^{152}Eu ^{153}Eu ^{154}Eu ^{155}Eu
^{152}Gd ^{154}Gd ^{155}Gd ^{156}Gd ^{157}Gd ^{158}Gd ^{160}Gd
^{165}Ho
^{169}Tm

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