Fall 2012

Hot channel determination and burnup analysis of Missouri University of Science and Technology Research Nuclear Reactor

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Department:

HOT CHANNEL DETERMINATION AND BURNUP ANALYSIS OF MISSOURI UNIVERSITY OF SCIENCE AND TECHNOLOGY RESEARCH NUCLEAR REACTOR

by

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A THESIS

Presented to the Faculty of the Graduate School of the

MISSOURI UNIVERSITY OF SCIENCE AND TECHNOLOGY

In Partial Fulfillment of the Requirements for the Degree

MASTER OF SCIENCE IN NUCLEAR ENGINEERING

2012

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ABSTRACT

A burnup analysis has been performed on the Missouri University of Science and Technology (Missouri S&T) Research Nuclear Reactor (MSTR). With use of the Monte Carlo neutronics depletion code MCNPX, burned material was input into a neutronics model (burned model) in order better simulate MSTR core characteristics. Simulated burnup values of ^{235}U for the past twenty years of MSTR operations totaled 14.266 grams, slightly less than the 14.527 grams reported by the reactor staff. A distribution of ^{235}U was simulated and a burnup map was developed.

Using the updated fuel material, the hot channel of the current configuration of the MSTR was determined by tallying energy deposition throughout the core. The hot channel is the limiting design factor of the core and is important in reactor safety analysis. The hot channel factor (ratio of hottest to average fuel plate) was determined to be 1.71 for the Core Configuration 120. Relative axial flux counts were simulated and experimentally measured. Simulated values were calculated with a maximum error of 8.54%compared to measured values. The burned model was tested to determine its improvement over the old model (clean model). In determining the effective multiplication factor at experimentally measured critical rod heights of the MSTR at 10 W, the burned model demonstrated a minimum 38% improvement over the clean model. The burned model was minimally 35% closer to simulating the experimentally measured shutdown margin and 46% closer to the excess reactivity. Incorporating burned material provided a more accurate model of the MSTR which can be in future core analysis.

ACKNOWLEDGEMENTS

I would like to sincerely thank my team of advisors, Dr. Ayodeji Alajo, Dr. Carlos Castaño, and Dr. Shoaib Usman. Working with three professors on a research project could have been challenging, but each contributed their own knowledge and was invaluable to me during this process. After every meeting I was motivated to get to work because of all of their enthusiasm.

I would like to thank Dr. Arvind Kumar for his support and his part in obtaining funding with the National Academy for Nuclear Training Fellowship and the Missouri S&T's Chancellor's Fellowship. The entire Nuclear Engineering department at Missouri S&T has been extremely supportive my educational development from an undergraduate through graduate school. I owe thanks as well to an integral part of the department, Sheila Johnson, without whom I would have been lost in the paperwork and deadlines.

I would like to thank my family and friends. Specifically I would like to thank fellow nuclear engineering graduate students Edwin Grant and Lucas Tucker for their patience and wisdom for my constant stream of questions. Finally, I would like to thank my mother, Gail O'Bryant. She is a constant source of strength and support in every aspect of my life including this work.

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1. INTRODUCTION

1.1. OBJECTIVE AND SCOPE

The nuclear engineering department at the Missouri University of Science and Technology (Missouri S&T) is supplemented by a nuclear research reactor, the Missouri University of Science and Technology Reactor (MSTR). The motivation of this work presented is to better understand current conditions in the MSTR. The department of nuclear engineering at the university has expressed interests in uprating the reactor to a higher power. An uprate would require a thorough investigation into the reactor physics of the MSTR by the reactor staff and the department and would need approval by the United States Nuclear Regulatory Commission (US-NRC). The US-NRC requires proof that existing systems can safely handle the increase in power (U.S. NRC). Reactor safety analyses must be performed in order to understand the reactor kinetics of the MSTR to ensure the reactor can handle the uprated power.

An increased power means reactor systems would be operating at higher temperature and would be at greater risk to fail. The limiting factor in the reactor system is at the location where the temperature is greatest. Heat is produced in the core of the MSTR and the hottest cooling channel in the core is known as the hot channel (Woodruff, 1997). By determining the hot channel of the MSTR, analyses can be performed with the limiting conditions known in order to determine the reactor's operational boundaries. The hot channel is expected to be near the center of the core where the neutron population is highest. This work aims to identify the hot channel through simulations using a Monte Carlo neutronics code, MCNP.

The MSTR has been using the same fuel elements since the reactor switched to low enriched uranium (LEU) fuel in 1992. In the twenty years of operation with the fuel, the reactor core has undergone several core configuration changes and has added fuel elements with the different configurations. The reactor is has not needed to refuel because it is not operated at high powers and is not operated continuously as is a commercial plant. The reactor staff records the amount of fuel burned every year, but does not have a method to track how much fuel is burned in each fuel element or plate. Through simulations, specific burnup amounts are assigned to each individual fuel plate. Using the same fuel for over twenty years leads to long lived neutron poison buildup which can cause a decrease in the reactor's excess reactivity. Presented is a method to track fuel burnup and poison buildup in the fuel plates of the MSTR through MCNPX simulation. Uneven burnup distribution can affect the reactor's reactivity and could change the hot channel location over time. By assessing the burnup of current fuel, and tracking energy deposition, an accurate model for determining the hot channel and simulating core characteristics accurately is developed.

1.2. MSTR

The MSTR is a pool type light water nuclear research reactor. It is used for educational purposes to train and teach students of Missouri S&T. The reactor is used for reactivity experiments, neutron activation analysis, radiation damage studies, neutron radiography and remote access to a high irradiation cell. The reactor facility contains a beam port, neutron tomography system, thermal column, and pneumatic transfer system to the reactor core. The MSTR is currently licensed at a power of 200 kW.

The MSTR consists of a fuel core in an open pool, attached to a bridge above the pool. The pool is light water which acts as a moderator, reflector, and shield for the reactor. A bulkhead separates a storage pit from the reactor core. The storage pit contains fuel elements not implemented in the core. A cooling system keeps the temperature of the water constant. The reactor system has a negative feedback coefficient for safety. The MSTR has recently incorporated a heat exchanger system which can be seen with the rest of the reactor set up in Figure 1.1 (Bonzer, William, 2009).

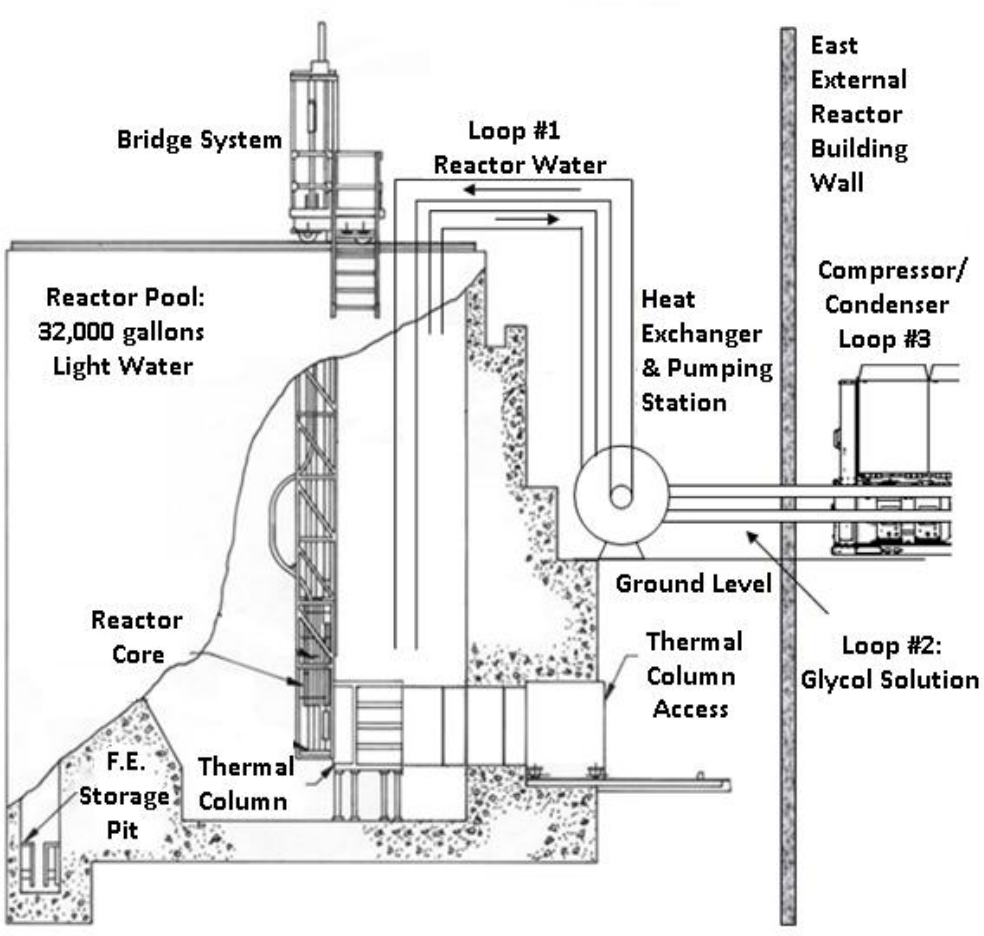


Figure 1.1. MSTR system schematic

1.3. REACTOR CORE

The MSTR core hangs in the reactor pool from a bridge over the pool. The base of the core is a grid plate system that allows for interchangeable components. The grid plate has a nine by six set of slots for those components. An ion chamber is level and adjacent to the core for detection purposes. Figure 1.2 is a picture of the MSTR core taken from the top of the pool.

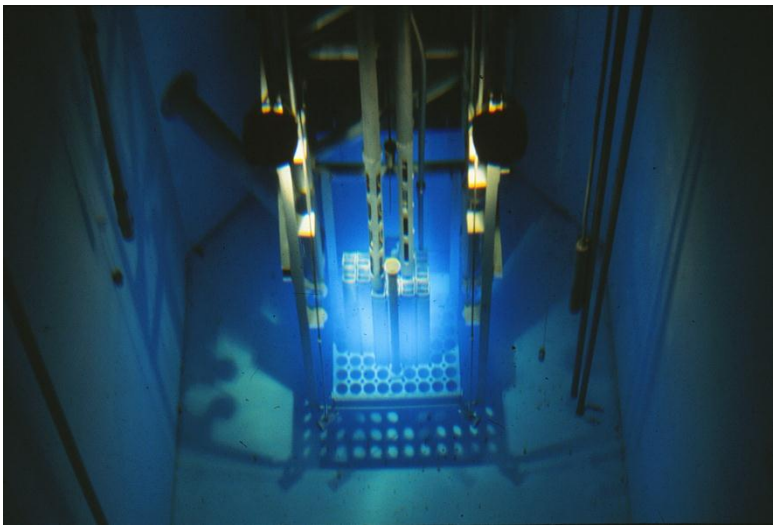
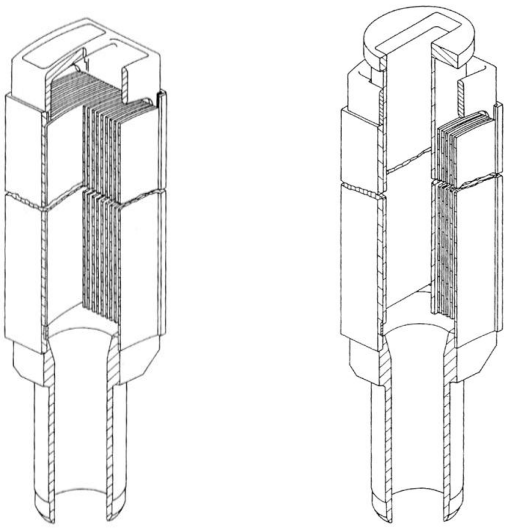


Figure 1.2. MSTR core

1.3.1. Source and Rabbit Tubes. The MSTR utilizes a Pu-Be source for startup. The source provides a minimum of two counts per second and is recorded on the reactor control room's startup channel. The bare and cadmium rabbit pneumatic tubes allow for quick irradiation of a sample while the high irradiation cell allows a user to irradiate a sample at high levels remotely.

1.3.2. Fuel Elements. Currently in Core Configuration 120, there are fifteen full fuel elements each contains eighteen LEU (<20% ^{235}U) curved fuel plates (Bonzer, William, 2009). The MSTR switched to LEU fuel because it is more proliferation resistant. The fuel plates are 0.051 cm slabs of U_3Si_2-Al (Uranium Silicide) surrounded by aluminum cladding 0.038 cm thick. Each plate is two feet long. Inside the full fuel element, the eighteen plates sit 0.4453 cm apart which is the gap that makes up a cooling channel. The MSTR uses control rods to introduce or remove negative reactivity to the core in order to control the criticality of the reactor. The reactor is equipped with three shim control rods composed of 1.5% natural Boron for course control and one regulating control rod made of stainless steel for fine control. Control rod elements contain ten fuel plates and a central slot for the rod. The MSTR has half fuel elements which contain nine fuel plates and room place samples for irradiation. From the MSTR Safety Analysis Report (SAR), Figure 1.3 shows a standard fuel element and also a control rod element (Bonzer, William, 2009).



(A) (B)

Figure 1.3. Fuel elements, (A) standard and (B) control rod

1.3.3. Configurations. The MSTR has changed core configurations several times for various reasons since the introduction of the current LEU fuel in 1992. The first configuration is titled Core Configuration 101. Core Configuration 101 was in place from July of 1992 until August of 2009 and is shown in Figure 1.4 below (Bonzer, William;, 1988).

	1	2	3	4	5	6	7	8	9
A									
B					S				
C				FE 8	FE 4	CR 4			
D			FE 13	CR 1	FE 3	FE 2	FE 12	FE 15	
E			FE 10	CR 2	FE 1	CR 3	FE 9	FE 14	
F			CRT	FE 5	FE 6	FE 7	BRT		

Figure 1.4. MSTR Core Configuration 101

Configuration 101 consisted of fifteen full fuel elements (FE), the four control rod elements (CR), the source (S), the bare rabbit tube (BRT), and the cadmium rabbit tube (CRT).

The excess reactivity of the MSTR decreased from 1992 to 2009 so to compensate, an additional half element (HE) was added to Core Configuration 118. Core Configuration 118 was in place from August 2009 until August of 2010 and is shown below in Figure 1.5 (Bonzer, William;, 1988).

	1	2	3	4	5	6	7	8	9
A									
B					S				
C				FE 8	FE 4	CR 4			
D			FE 13	CR 1	FE 3	FE 2	FE 12	FE 15	
E			FE 10	CR 2	FE 1	CR 3	FE 9	FE 14	HE 1
F			CRT	FE 5	FE 6	FE 7	BRT		

Figure 1.5. MSTR Core Configuration 118

The half element was removed and an additional full element was added to Core Configuration 120 and the core was moved closer to the thermal column to achieve a higher neutron flux there. Additionally, the high irradiation cell (HC) was included. Figure 1.6 shows Core Configuration 120, which is the current configuration.

	1	2	3	4	5	6	7	8	9
A									
B						S			
C					CR 4	FE 5	FE 1	FE 17	
D				FE 4	FE 8	FE 14	CR 1	FE 10	FE 2
E				FE 9	CR 3	FE 12	CR 2	FE 7	FE 3
F				CRT	FE 15	HC	FE 13	BRT	FE 6

Figure 1.6. MSTR Core Configuration 120

Each core configuration has different reactivity and neutron flux characteristics. Individual fuel elements will burn fuel differently in each configuration. Each configuration may have a unique hot channel, but the primary concern is identifying the hot channel for Core Configuration 120, as it is the current configuration.

1.4. MCNP MODELING

MCNP is a computer program developed by Los Alamos National Lab for simulating neutron environments (Los Alamos National Laboratory, 2008). The program uses a Monte Carlo method for modeling systems. The input for the program is a description of the geometry of the environment using surfaces to define cells and material definitions for what those cells consist of. The surfaces are infinite planes or cylinders, spheres, or some macro bodies such as toroids or parallelepipeds. The cells are then defined as some combination of these surfaces with respect to which side of the surface. The cell is then defined as consisting of some material at some density. The material is defined as containing some particular mixture of isotopes.

A definition of the source of particles is also required. This definition includes where, with what energy, and headed in what direction each particle starts and with what frequency the particles are started with those characteristics. Finally a description of desired output from the program is defined. This usually consists of a number of tallies for determining flux values or energy deposition at particular locations. The program then uses a library of energy dependent cross sections and the material definitions to calculate macroscopic cross sections for all possible particle interactions within each cell.

The particles are started according to the source definition and tracked through the system. The particle undergoes interactions according to probabilities determined from the cross sections calculated from the material definitions. The particle is tracked until it has an absorption or fission interaction or leaves the system.

The program may also be run in "kcode," so that the effective multiplication factor (k_{eff}) of the system may be calculated. The program starts the particles in cycles. For the first cycle the particles are started according to the defined source. For each subsequent cycle the particles are started at the sites where fission interactions occurred in the previous cycle. kcode was run with 20000 histories and 1015 cycles. The k_{eff} of the system is then the number of neutrons produced from fission divided by the number of neutrons started in the cycle.

A model of the MSTR was originally created by Dr. Jeffery King (King, 2008), which was updated for the current geometry and core configurations. Additional geometries changes were required in order to track energy deposition in each fuel during simulation. MCNPX version 2.6 was used for burnup analysis. MCNPX requires additional information such as power, length and number of time steps, materials, power

fractions and which fission products to inventory. The second tier of fission products was selected for the burnup analysis because this tier includes significant poisons that could affect the reactivity of the MSTR. This list of isotopes can be seen in Table B.1. MCNP runs a set of ten statistical tests as part validation of tally values and all values obtained through simulations passed all statistical tests. Figure 1.7 shows the MCNP geometry of the MSTR pool system. Figure 1.8 is an overhead view the MCNP MSTR Core Configuration 120 core. Figure 1.9 shows the axial and radial directions in reference to the MSTR core (Bonzer, William, 2009).

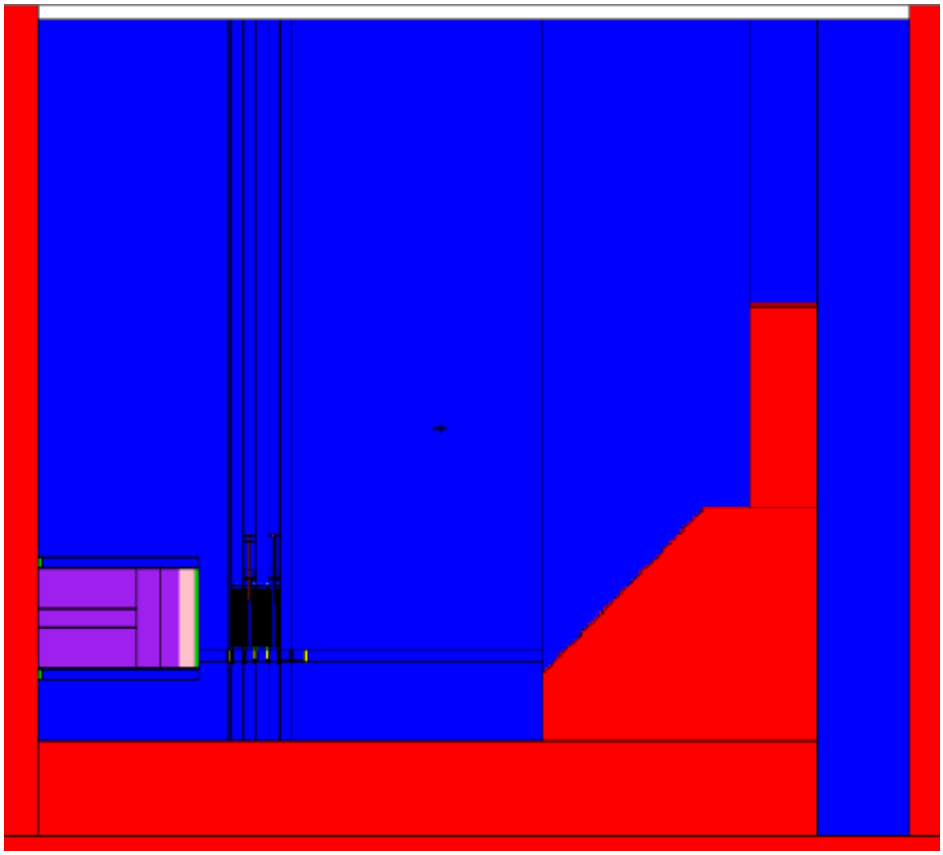


Figure 1.7. MCNP MSTR pool profile view

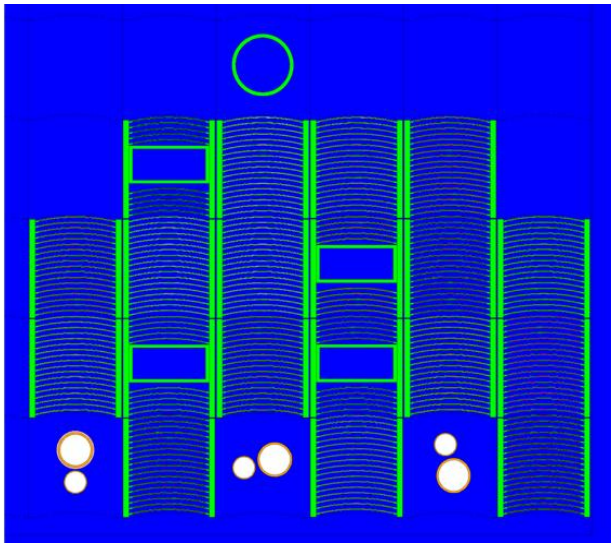


Figure 1.8. MCNP overhead view of MSTR Core Configuration 120

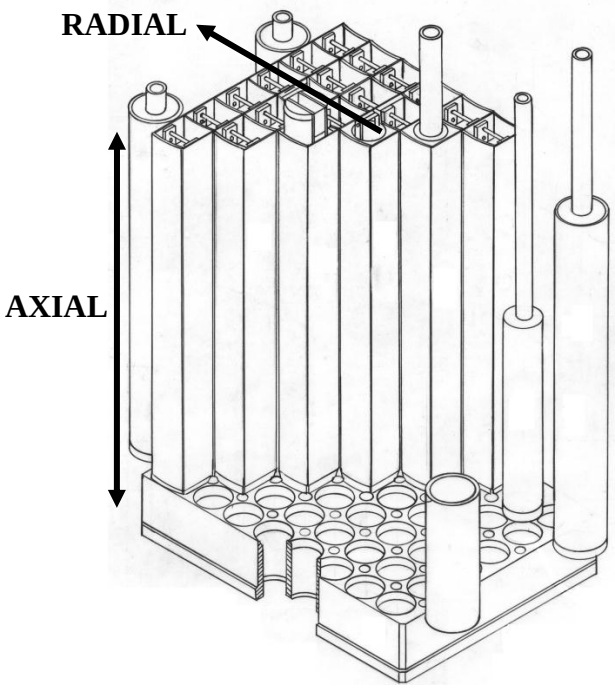


Figure 1.9. Reference directions of MSTR core

2. HOT CHANNEL DETERMINATION

Containment of fissile material is a primary concern and thus much emphasis is put into the prevention of fuel cladding failure. Power peaking occurs in the hot channel of a reactor where the heat flux is at a maximum. In a reactor core, the hot channel is the limiting factor in design constraints. If the hot channel structural material is deemed safe, the remaining channels will be safe too because they operate in lower temperatures. When performing safety analysis it is essential to locate the reactor's hot channel in order to analyze the most vulnerable section of the reactor core. The hot channel factor relates maximum to average design conditions (Glasstone, 1994).

2.1. NEUTRON EXTRAPOLATION DISTANCE

Geometry is an important factor in determining the hot channel of a nuclear core. The distance to other plates, moderators and reflectors from a fuel plate impact the neutron utilization in that plate. Understanding the neutron flux in the radial direction of the core will yield a better idea of how relative distances in the geometry affect neutron then position of the hot channel in that configuration. Equations 1 and 2 demonstrate how the extrapolation distance, d, was determined for the light water in the MSTR core (Duderstadt, 1976).

$$\boldsymbol{d = 2.13D} \tag{1}$$

$$\boldsymbol{D = \frac{\lambda_{TR}}{3}} \tag{2}$$

Where D is the diffusion length and λ_{TR} is the transport mean free path of water.

The distance between fuel plates is .4453 cm and the extrapolation distance was determined to be 1.63 cm. This means that the extrapolated distances from one plate covers two plates and nearly three cooling channels below it. Figure 2.1 is a close up view of a fuel element showing the cooling channel width (y) and extrapolation distance (d) of the top fuel plate.

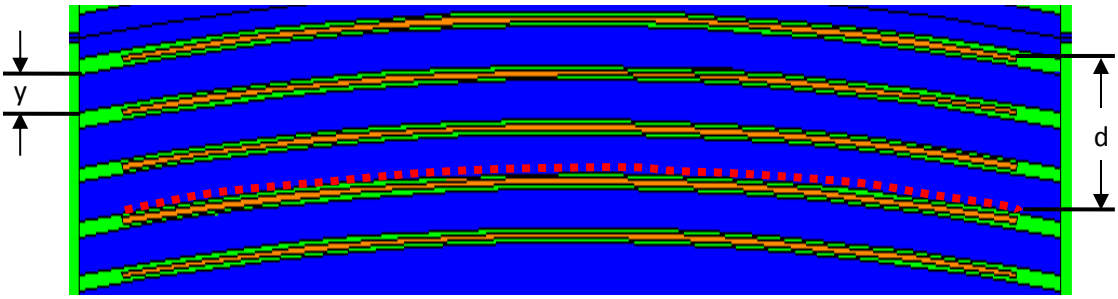


Figure 2.1. Cooling channel width and extrapolation distances

2.2. HOT CHANNEL METHODOLOGY

Commercial reactors use sub channel analysis and various system codes in order to determine the hot channel. The MSTR is a research reactor and has a non-uniform core configuration and unique operating conditions. In order to determine the hot channel, MCNP was used to track energy deposition in each fuel element of the core.

The MCNP5 model used to in this simulation was validated with axial flux measurements and the approach to criticality technique (Richardson, 2011). Several assumptions were made in the model simulation to determine the hot channel. Namely, a clean, isothermal core was initially assumed with no poison buildup. The decision to neglect fuel burnup was based on the extremely low reported burnup (from 1984-2004 is

less than 10 MW-hr per year (MSTR, LEU Fuel Burnup Conversion, 2012)). An F6 tally was used in the MCNP5 code; an MCNP5 manual states that "true heating is found by summing the neutron and photon F6 tallies in a coupled neutron/photon calculation" (Los Alamos National Laboratory, 2008).

2.3. CLEAN CORE HOT CHANNEL RESULTS

Energy deposition in each fuel plate of Core Configuration 101 was simulated with a MCNP model of a critical MSTR at 200 kW. The results were mapped and normalized so that the percent contribution from each plate is seen in Figure 2.2. Only core positions that contain fuel plates are shown.

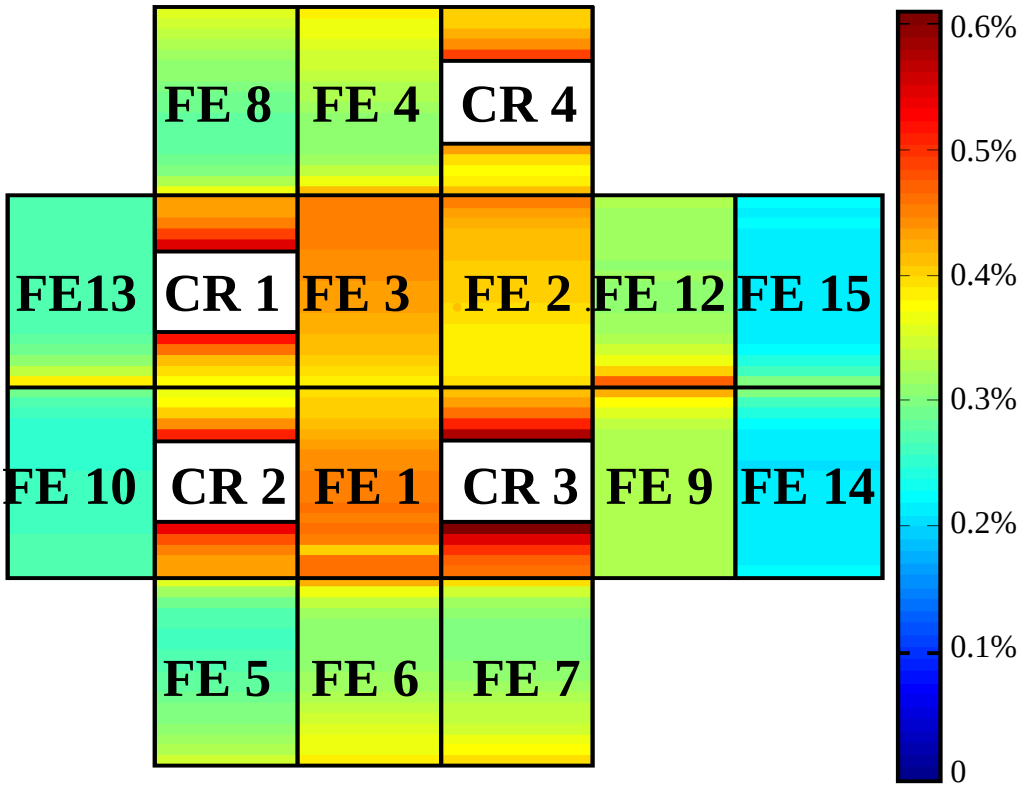


Figure 2.2. Fractional energy deposition in fuel plates, Core Configuration 101

The average fuel plate contributed 0.342% to the total energy deposition. The hottest plate contributed 0.608% of the energy deposition and the maximum to average ratio is 1.778. The hot channel of this channel corresponds to the channel between the two plates with the highest combined energy production. In a clean Core Configuration 101, the hot channel is located between fuel plates four and five of control rod element three. The simulation was repeated with a clean Core Configuration 120 to compare energy deposition distribution and is seen in Figure 2.3.

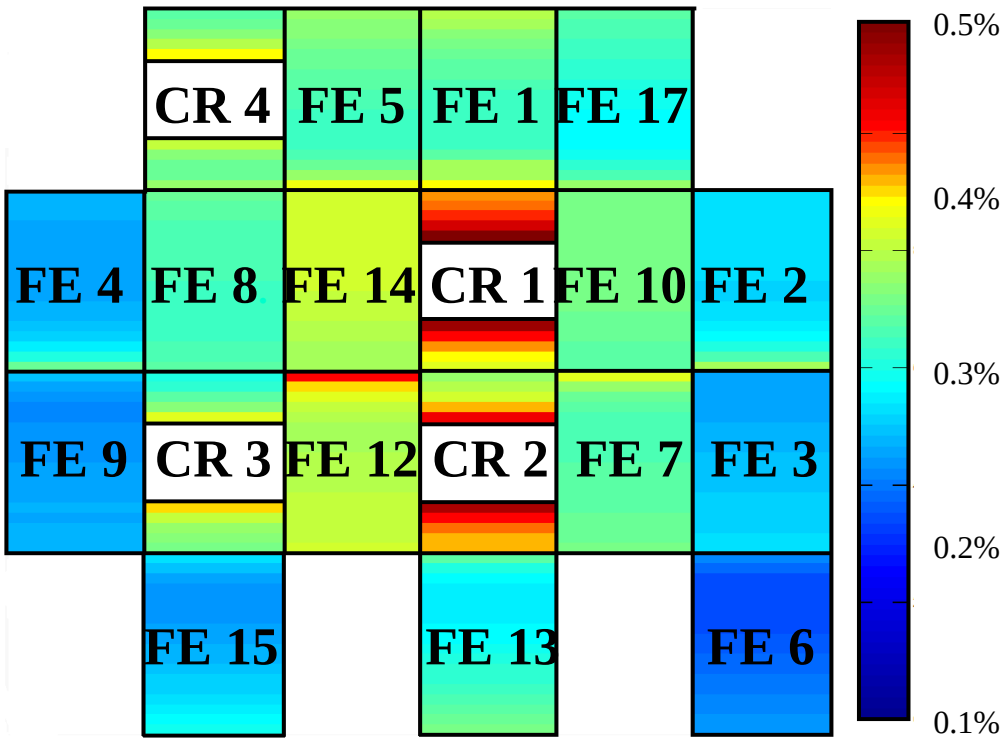


Figure 2.3. Fractional energy deposition fuel plates, Core Configuration 120

The simulation tallies energy deposition averaged over each fuel cell. A total of 310 fuel plates are represented in Figure 2.3. As expected, neutron utilization is greater in the center of the core where the neutron flux is maximum, which lead to higher energy deposition. Each fuel plate, on average, contributes 0.323% of the total energy deposition. The hottest plate contributes 0.597% of the total energy deposition which yields a factor of about 1.85 times greater than the average plate.

In both simulations the two fuel plates with the highest energy deposition are in the fuel elements of control rods near the center of the core configuration. The reason for the hottest channel being next to the control rod is because when the reactor is critical; the control rods are withdrawn so that only ten centimeters of the space in the top of the core is occupied by the control rod. The rest of the space is filled with water which acts as a moderator and reflector producing better thermalization of the neutrons and more energy deposition in the fuel plates adjacent to the control rods. In order to corroborate this finding, an additional simulation was run exchanging the material next to the to the control rod 1 with thin natural Cadmium. The contribution to the total energy deposition in the Cadmium plate decreased to 0.122% (Originally 0.5967%) and the energy deposition in other fuel plates in the same fuel element drastically decreased as well which agrees with previous findings.

2.4. CLEAN CORE HOT CHANNEL CONCLUSIONS

The determination of the hot channel of the MSTR 101 and 120 cores was based on several assumptions. The zero burnup assumption was made because of the relatively

low power at which the reactor is licensed to operate. However, the energy deposition was determined to be significantly higher in the fuel plates of the hot channel than the average value, meaning power peaking could occur and higher burnup exists in those plates. The current fuel in the MSTR has been in use since 1992. The hot channel analysis may be sufficient for the beginning of operations for Core Configuration 101, but burnup in the hot channel fuel plates would decrease the reactivity of the core over time. Because of the uneven burnup, the hot channel may have shifted to a different location in the core. A burnup reconstruction is required to determine the location of the hot channel for current Core Configuration 120.

3. BURNUP ANALYSIS

MCNPX was used to track fuel burnup in the fuel plates over the lifetime of the current fuel. MCNPX requires a burn card in addition to the MCNP code which includes values such as power, time steps, power fraction, fission product tracking tier, material numbers, and material volumes. The second tier of fission products was chosen because the tier contains isotopes with relevant half-lives and cross-sections. Each fuel element contains eighteen fuel plates that initially were composed of the same material, but as seen in the hot channel analysis, the energy deposition varied greatly among plates in the same element. Burnup also would vary as well, so in this analysis, it was important to track the burnup of each individual fuel plate.

A key assumption made in this analysis is that the burnup simulation averages the burned material over the entire cell burned. In the MSTR model the neutron flux varies greatly in the axial direction but the burnup is averaged over that direction as the cell being burned is the entire fuel plate. In future work each plate can be divided axially into multiple cell segments to account for uneven burn in the axial direction of the fuel plates.

MCNPX runs an auto-corrector subprogram for kcode for each time step. As such, computer time is heavily dependent on the number of time steps in the simulation. Each new power is associated with a new time step. This presents a problem in simulating the burnup of the MSTR because it is operated randomly at various power levels throughout a single day. Running simulations for a single day takes longer than a single day to run, meaning simulating twenty years with exact powers would take longer than twenty years. This is not a practical analysis, to cut down on computer time an

equivalent averaged power is used over longer time steps. An analysis for optimizing this number of time steps was performed and is presented in Section 3.1.

3.1. EQUIVALENT AVERAGED POWER JUSTIFICATION

In order to determine if it is an accurate assumption to average powers in the MCNPX burnup simulations, a study was done to demonstrate the effects of averaging power over different time steps. Four different studies were done, comparing actual power to a single day time step, seven one-day time steps to a single week-long time step, four week-long time steps to a single month long time step, and finally twelve month-long time steps to a single year long time step. The goal of these studies is to justify averaging power in order to use as few time steps as possible in the burnup simulations while maintaining an accurate fuel makeup.

3.1.1. Actual Power Compared to Single Day Equivalence. This study will compare differences in fuel makeup between a simulation following actual power levels throughout a twenty-four hour period to a simulation using one time step of a day with an averaged equivalent power over that twenty-four hour period.

An average reactor operating day was chosen from the power logbook. First, an MCNPX model was run to simulate the day exactly at the power and times the reactor actually operated on that date. Fission products buildup and fuel burnup were tracked and are provided at the end of each time step (whenever the reactor changed powers). Next, the total energy of the day was averaged over twenty-four hours in an MCNPX

model with a single time step at the average power run. Figure 3.1 shows the powers of the two different simulations throughout the same day.

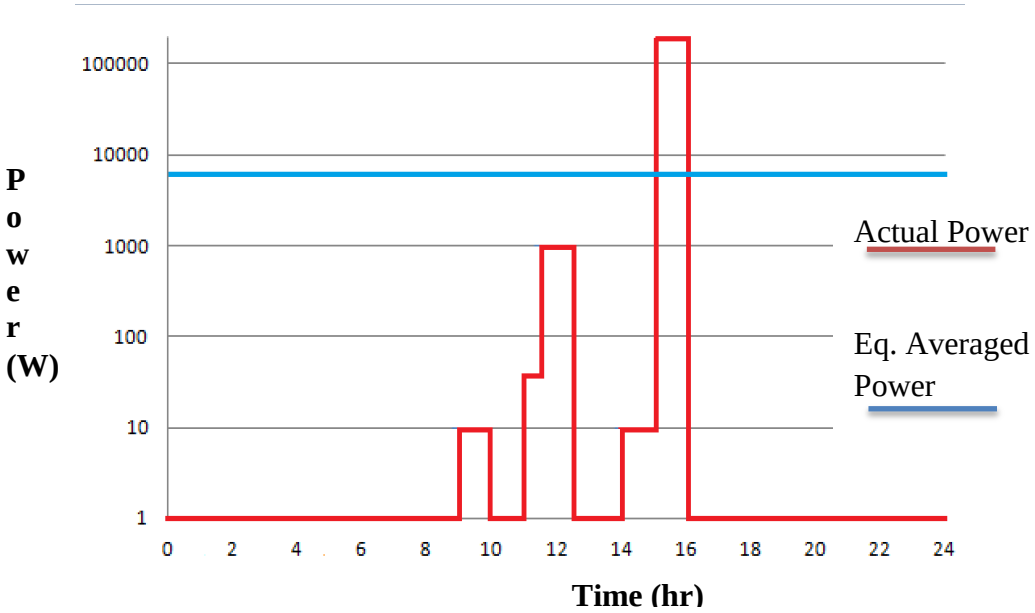


Figure 3.1. MSTR actual operational power and equivalent averaged power

It is important to note that the power on the y-axis of Figure 3.1 is set as a log scale due to the wide range of actual power operation throughout the given day. The key to averaging is to ensure that the energy spent over the time period (one day for this study) is the same for the averaged simulation as for the actual operation simulation. Table 3.1 shows the power levels over the day and the energy generated through the actual operating conditions.

Table 3.1. MSTR operational power and control rod heights (08/23/1992)

Time	Power (W)	CR Heights (in)	RR Height (in)	Energy (W-day)
0:00	0.00	0.00	0.00	0.00
9:00	10.00	19.30	13.70	0.417
10:00	0.00	0.00	0.00	0.00
11:00	40.00	20.75	15.50	0.83
11:30	1000	21.00	14.30	41.67
12:30	0.00	0.00	0.00	0.00
14:00	10.00	19.30	13.90	0.417
15:00	200000	21.40	14.00	8333.33
16:00	0.00	0.00	0.00	0.00
24:00:00	0.00	0.00	0.00	0.00
			Total	8376.67

Table 3.1 shows the three control rod heights (CR Heights) and the regulating rod height (RR Height). These values correspond to the amount of the rod removed. For example, at a power of zero watts all rods are inserted or are at a height of 0 inches removed. Critical rod heights change at different powers due to changes in material properties that occur because of temperature change. The reactor does not have set critical rod heights for each power setting; rather, different combinations of rod heights can be used to achieve criticality. In the simulation of the actual power operations those rod heights are changed in the geometry. However, for the averaged power simulation critical rod heights are not predetermined. Once the averaged equivalent power was determined to be 8376.67 watts, recorded critical rod heights for a similar power were used in the averaged simulation. These heights were interpolated from data in the MSTR power logbook (Reisner, 2012) for a near clean, cold reactor core. The control rods were set to 21.00 inches removed and the regulating rod was set to 15.60 inches removed.

The results of the study are summarized in Table 3.2 and Table 3.3. Table 3.2 shows the actinide inventory for both simulations at the beginning (initial) and end of the day (Final) of operation while Table 3.3 lists the non-actinide inventory. These tables show the inventories for the fuel plate which borders the predicted hot channel for the cold, clean, Core Configuration 101.

Table 3.2. Actinide inventory for day long study

	Mass (g)		
Isotope	Initial	Final Actual	Final Avg'd
U 238	2.81E+00	2.81E+00	2.81E+00
U 235	6.94E-01	6.94E-01	6.94E-01
U 236	0.00E+00	7.43E-06	4.67E-06
Np 239	0.00E+00	0.00E+00	0.00E+00
U 239	0.00E+00	1.87E-06	1.87E-06
Pu 239	0.00E+00	0.00E+00	0.00E+00
U 237	0.00E+00	0.00E+00	0.00E+00

Table 3.3. Non-actinide inventory for day long Study

	Mass (g)		
Isotope	Initial	Final Actual	Final Avg'd
Al 27	1.807E+00	1.807E+00	1.807E+00
Si 28	2.536E-01	2.536E-01	2.536E-01
Si 29	1.333E-02	1.333E-02	1.333E-02
Si 30	9.092E-03	9.092E-03	9.092E-03

The inventories show no difference in fuel (^{235}U) burnup from the initial and final time steps. For only one day of operation, the MSTR does not burn enough fuel to indicate a significant difference in the MCPNX simulation inventory as seen in Table 3.2. The important fact to take out of these results is that there is no difference in fuel burnup between the two simulations. Long term poison buildup is a concern as well, so tracking isotopes with large neutron absorption cross-sections and long half-lives is important. The amount of poison buildup after one day is negligible. Thusly, averaged equivalent power can be used to simulate one day without having significant effects on the final inventory of the fuel. The computer time required to simulate the power precisely was 76 hours, compared to 13.125 hours for the averaged equivalent power.

3.1.2. Four Week to Single Month Comparison. A second comparison was conducted using the same approach, except comparing seven one-day time steps and one week long time step. The inventory again did not yield any burnup or differences in between the two simulations so it was not included. Another time step compared four week-long time steps to a single month-long time step of averaged equivalent power (averaged over 28 actual operating days). An average week of operating at the MSTR is equivalent to operating continuously at a power of 5982 watts. The control rods were again set to an equivalent rod height for this power. The inventory results of the four time steps (Four Weeks) and the one time step (Month) are summarized in Table 3.4 and Table 3.5.

Table 3.4 Actinide inventory for month long study

	Mass (g)		
Isotope	Initial	Final, Four Week	Final, Month
U 238	2.81E+00	2.81E+00	2.81E+00
U 235	6.94E-01	6.94E-01	6.94E-01
U 236	0.00E+00	1.46E-04	4.67E-06
Np 239	0.00E+00	0.00E+00	0.00E+00
U 239	0.00E+00	0.00E+00	1.87E-06
Pu 239	0.00E+00	0.00E+00	0.00E+00
U 237	0.00E+00	0.00E+00	0.00E+00

Table 3.5. Non-actinide inventory for month long study

	Mass (g)		
Isotope	Initial	Final, Four Week	Final, Month
Al 27	1.81E+00	1.81E+00	1.81E+00
Si 28	2.54E-01	2.54E-01	2.54E-01
Si 29	1.33E+00	1.33E-02	1.33E-02
Si 30	9.09E-03	9.09E-03	9.09E-03
Xe 136	0.00E+00	3.14E-05	3.22E-05
Cs 137	0.00E+00	2.81E-05	2.87E-05
Ba 138	0.00E+00	3.07E-05	3.16E-05
Zr 96	0.00E+00	2.02E-05	2.06E-05
Zr 94	0.00E+00	2.01E-05	2.06E-05
Ru 101	0.00E+00	1.73E-05	1.77E-05
Xe 135	0.00E+00	5.182E-07	5.24E-07
Nd 145	0.00E+00	1.865E-05	1.91E-05
Zr 92	0.00E+00	1.809E-05	1.85E-05
Zr 93	0.00E+00	1.909E-05	1.95E-05
Cs 135	0.00E+00	2.539E-05	2.59E-05
Cs 133	0.00E+00	2.023E-05	2.08E-05
Nd 143	0.00E+00	1.170E-05	1.20E-05
Tc 99	0.00E+00	1.696E-05	1.74E-05
Sm 149	0.00E+00	1.49E-08	1.131E-08
Nd 143	0.00E+00	1.170E-05	1.20E-05

The inventory results show no difference in ^{235}U burnup, but do show a slight difference in the buildup of ^{236}U. The non-actinide inventory a slight difference in ^{135}Xe and other isotopes, but the mass values are so small that they are negligible. Again differences in inventories of the two simulations is negligible and averaging power over month took thirty hours less time than simulating four weeks in four time steps.

3.1.3. Year Long Study. A fourth and final comparison was made between inventories of a twelve month-long time step simulation and a one year-long time step simulation. The entire first year of operation of the MSTR is equivalent to operating continuously at 5752 watts for one year. The three control rods and the regulating rod were placed in the appropriate withdrawal heights in the MCNPX model geometry. The actinide and non-actinide inventories are shown in Table 3.6 and Table 3.7, respectively.

Table 3.6. Actinide inventory for year long study

	Mass (g)		
Isotope	Initial	Final, 12Mo	Final Avg'd
U 238	2.811E+00	2.810E+00	2.810E+00
U 235	6.937E-01	6.827E-01	6.821E-01
U 236	0.000E+00	1.741E-03	1.842E-03
Np 239	0.000E+00	0.000E+00	0.000E+00
U 239	0.000E+00	0.000E+00	0.000E+00
Pu 239	0.000E+00	7.851E-04	8.333E-04
U 237	0.000E+00	0.000E+00	0.000E+00

Table 3.7. Non-actinide inventory for year long study

	Mass (g)		
Isotope	Initial	Final, 12Mo	Final, Year
Al 27	1.81E+00	1.807E+00	1.807E+00
Si 28	2.536E-01	2.536E-01	2.536E-01
Si 29	1.333E-02	1.334E-02	1.33E-02
Si 30	9.09E-03	9.092E-03	9.092E-03
Xe 136	0.00E+00	3.939E-04	3.97E-04
Cs 137	0.00E+00	3.483E-04	3.50E-04
Ba 138	0.00E+00	3.876E-04	3.90E-04
Zr 96	0.00E+00	2.528E-04	2.54E-04
Zr 94	0.00E+00	2.525E-04	2.54E-04
Ru 101	0.00E+00	2.175E-04	2.19E-04
Xe 135	0.00E+00	2.968E-07	4.95E-07
Nd 145	0.00E+00	2.341E-04	2.38E-04
Zr 92	0.00E+00	2.272E-04	2.31E-04
Zr 93	0.00E+00	2.448E-04	2.46E-04
Cs 135	0.00E+00	3.202E-04	3.31E-04
Cs 133	0.00E+00	3.645E-04	3.63E-04
Nd 143	0.00E+00	3.386E-04	3.33E-04
Sm 149	0.00E+00	3.127E-05	3.11E-05
Tc 99	0.00E+00	2.471E-04	2.50E-04
Nd 143	0.00E+00	3.386E-04	3.33E-04

These results show that averaging power over the time period of a year does not significantly change the inventory of the fuel at the final time step. The computer time needed for the twelve month-long time step simulation was 135.5 hours (5.65 days) compared to 18 hours for the equivalent averaged power simulation. By averaging power over longer periods of time significant computer time is not only saved, the twenty-year

burnup analysis is made practical. These studies show the practicality in using equivalent averaged power levels.

3.1.4. Equivalent Averaged Power Conclusions. The four studies indicate that in order to determine the fuel burnup, poison buildup and final makeup of the MSTR fuel, using equivalent averaged power will yield inventories comparable to simulating exact powers but save significant computer time. Therefore, simulations with time steps of one year were implemented in the burnup analysis from 1992 to 2012. Equivalent power was averaged using values from the MSTR power logbooks and the rod heights were set according to that year's equivalent averaged power. An example of the logbook can be seen in Table A.1.

3.2. POISON BUILDUP

Due to computational limits, simulation of the exact power history of the MSTR is not practical. In order to apply an equivalent averaged power to the computer simulated model it is necessary to assess if the equivalent averaged power simulation has the same effect on poison buildup as when the reactor is actually operated. A study was simulated in order to gauge the effects of poison buildup under actual operating conditions

3.2.1. ^{135}Xe Poisoning. The MSTR is not operated as a traditional utility reactor in that it can be started up and shut down numerous times throughout a single day of operation. Some neutron absorbing isotopes (poisons) such as Xenon-135 are produced at high rates immediately after the reactor is shutdown. The fission product ^{135}I beta decays

to ^{135}Xe, which eventually decays away with a half-life of 9.14 hours (Knolls Atomic Power Lab, 2010). When a reactor is shutdown, the ^{135}I continues to decay to ^{135}Xe which causes the amount of ^{135}Xe to follow according to equation 3 (Lamarsh, 2001).

$$\boldsymbol{X(t) = X_o e^{-\lambda_X t} + \frac{\lambda_I I_o}{\lambda_I - \lambda_X}(e^{-\lambda_X t} - e^{-\lambda_I t})} \quad (3)$$

Where X(t) is the isotopic concentration at time t, X_o is the initial isotopic concentration, and λ is the isotopic decay constant. ^{135}Xe has a large cross section (2.943 megabarns) (Knolls Atomic Power Lab, 2010) and thus can introduce enough negative reactivity short term into a reactor core to keep the reactor from being able to achieve criticality. This time period where reactor cannot achieve criticality is known as reactor deadtime. If ^{135}Xe buildup prohibited the ability of the MSTR to reach criticality, it would be reflected in the power logbooks and hence accounted for simulating with an equivalent averaged power over time. However, if enough ^{135}Xe concentrates in the hot spots of the core it may cause a shift in where the reactor fuel burns and thus changes the burnup distribution. In order to validate the assumption which ignores startup and shutdowns of the reactor, it is necessary to confirm that the reaction rate from the ^{135}Xe produced is negligible compared to the reaction rate with the fuel, ^{235}U, present in the hot channel. If ^{135}Xe is determined to be negligible in the hottest channel, short term poisons will be disregarded because ^{135}Xe is the largest threat to the assumption (Bonzer, William, 2009). The reaction rate is given by equation 4 (Shultis, 2008).

$$\boldsymbol{RR = \sigma \phi N} \quad (4)$$

Where *RR* is the reaction rate, σ is the neutron absorption cross-section, ϕ is the neutron flux, and N is the isotopic concentration. In order to obtain the most

conservative comparison of the reaction rates of ^{135}Xe and ^{235}U, it is necessary to determine the maximum amount of ^{135}Xe present. By solving equation 3 for t when X(t) is at a maximum, equation 5 (Duderstadt, 1976) is obtained.

$$\boldsymbol{t_{max} = \frac{1}{\lambda_I - \lambda_X} \ln\left[\frac{\lambda_I}{\lambda_X}\right] = 11.6\ hr} \tag{5}$$

The highest concentration of ^{135}Xe that will accrue 11.6 hours after the reactor is shutdown from its maximum power, which is 200 kW. Figure 3.2 (Duderstadt, 1976) depicts ^{135}Xe buildup to steady state during a reactor operation of a traditional reactor and then its peak and eventual decay after shutdown at t_o.

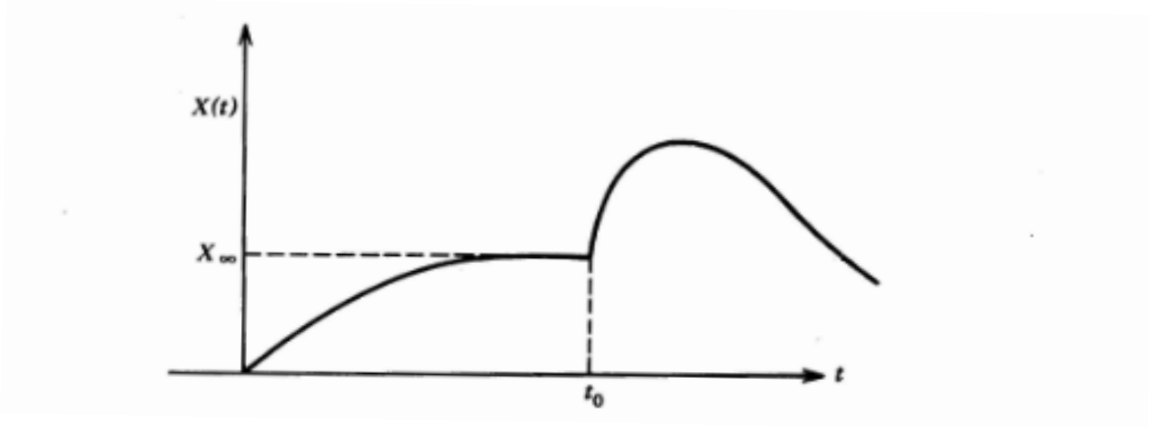


Figure 3.2. Qualitative behavior of ^{135}Xe concentrations following a cold, clean startup and then shutdown

3.2.2. ^{135}Xe Poisoning Results. The MSTR is only operated for eight hours at a time at most. A MCNPX simulation was set up to track the buildup of ^{135}Xe during an eight hour run at 200 kW and twenty-two hours following the reactor shutdown. Figure 3.3 shows the buildup and decay of 135Xe over that time with data points at every half of an hour and an additional data point at the maximum concentration time of 11.6 hours after shutdown.

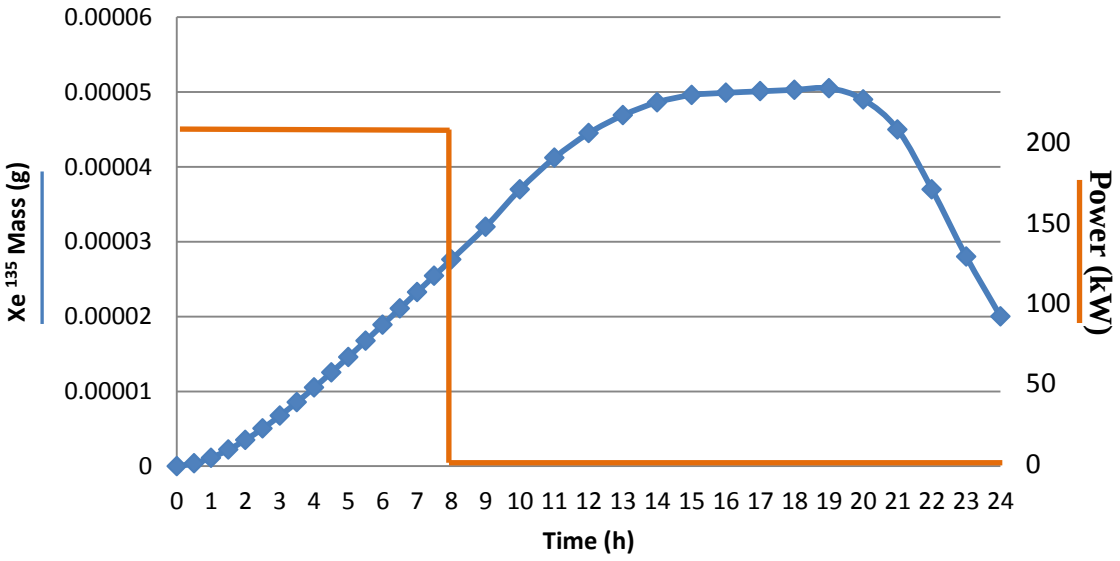


Figure 3.3. ^{135}Xe buildup in MSTR during eight hour operation at 200 kW and after shutdown

In Figure 3.3, the MSTR was not operated long enough for the poison to reach equilibrium (^{135}Xe decays away as fast as ^{135}I decays into ^{135}Xe). The maximum amount of ^{135}Xe occurs about 11.6 hours after shut down as predicted by equation 5 which occurs at 19.6 hours in Figure 3.3. The mass and number of atoms of both ^{135}Xe and ^{235}U are

shown in Table 3.8 along with the reaction rates at that time according to equation 4. The number of atoms present is the product of the isotope's molar weight and the mass in grams.

Table 3.8. Reaction rates for ^{135}Xe and ^{235}U 11.6 hours after MSTR shutdown

Isotope	Mass (g)	Molar Mass (g/mol)	Atoms (Mol)	σ (b)	RR (Mol*b)
^{135}Xe	5.01-05	134.9072	3.54E-07	2.94E+6	9.65E-01
^{235}U	2.05E+01	235.04392	8.74E-02	6.98E+2	5.13E+01

Dividing the reaction rates shows that the reaction rate of ^{135}Xe is only about 1.88% of the reaction rate of ^{235}U. This is sufficiently small that the effects of ^{135}Xe and short term poisons can be neglected and the burnup simulations can use equivalent averaged power without concern of effects by ^{135}Xe.

3.3. BURNUP

Three core configurations were considered in the burnup analysis; Core Configurations 101, 118 and 120. All other configurations were not implemented long enough relative to the twenty year time frame of the analysis. The reactor can be operated in "T-mode" where the bridge and core are moved closer to the wall to provide a higher neutron flux for the thermal column, or "W-mode" where the bridge and core are backed away from the wall. For the burnup analysis it was assumed the reactor was in "W-mode" in order to decrease the number of time steps necessary and thus decrease total computation time.

After each time step, the new material inventory for each fuel plate is updated for the next time step. For example, if after the first year of simulated operation 0.0001 grams of ^{235}U were burned from fuel plate one of fuel element fourteen, then for the second year's simulation that loss of 0.0001 grams would be incorporated into the material card of that fuel plate. The fuel plates are assumed to be homogenous; this model does not take into account variations in axial burnup.

3.3.1. Burnup Results. After the MSTR burn simulation brought the fuel plate material inventory up to date (8/564/2012), the ^{235}U burnup was calculated for each fuel plate. The mass of ^{235}U from the inventory of the final time step was subtracted from the initial mass. The results are mapped in Figure 3.4 below.

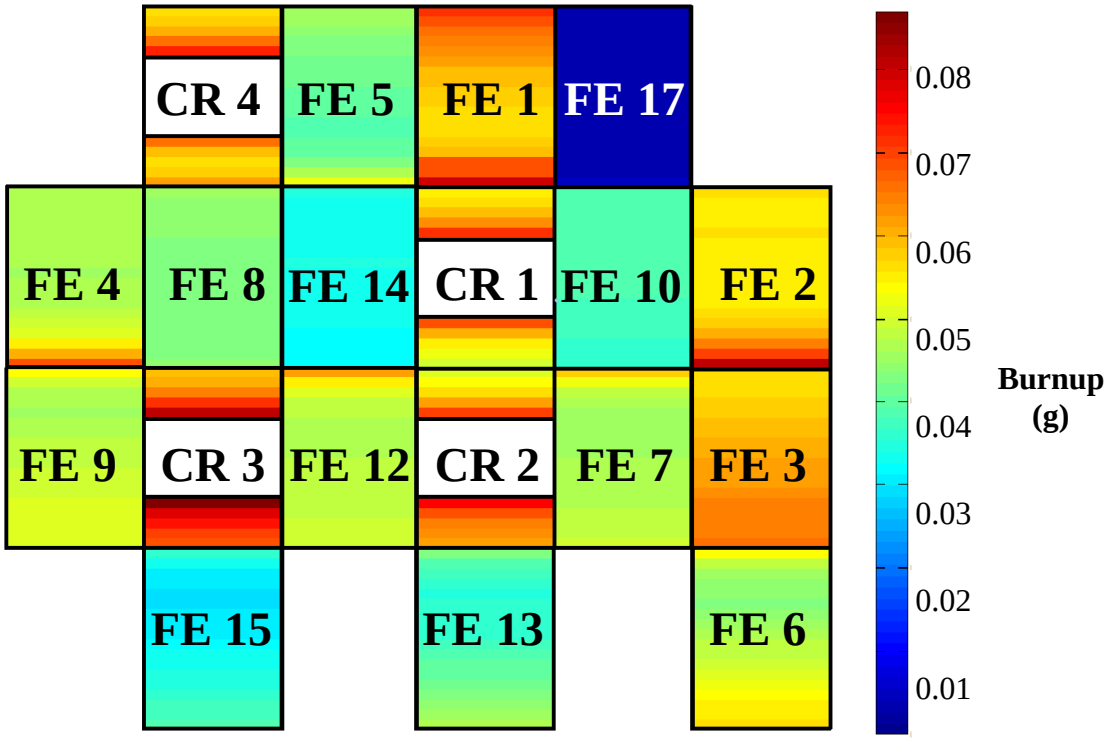


Figure 3.4. ^{235}U fuel plate burnup map, Core Configuration 120

As was expected, the most burnup occurred in the center of the core corresponding to the highest neutron utilization. The burnup map of Figure 3.4 closely resembles the map of fractional energy deposition in fuel elements (Figure 2.2) which again makes sense because it is the fission of ^{235}U that causes the energy deposition. Fuel element seventeen has significantly lower burnup than the rest of the fuel elements because it was only introduced in the core for the past two years in Configuration 120.

The plate with the highest ^{235}U burnup is in control rod element three, plate five which is adjacent to the gap for the control rod. This control rod element was in the center of Core Configuration 101 which was used for eighteen of the twenty years of this burnup analysis. The energy deposition for Core Configuration 101 shows the hot channel is bordered by this plate. The plate's proximity to a strong reflector and location in the center of the configuration led to higher neutron utilization and, as a result, higher burnup.

3.3.2. MSTR ^{235}U Distribution Comparison. The MSTR staff log the operating power of the reactor and calculate how much ^{235}U is burned yearly based on how much energy is produced by the core (energy generated is the nominal power logged multiplied by the time at power). The staff reports to the US-NRC how much fuel is burned and assign the burnup to certain fuel elements (MSTR, LEU Fuel Burnup Conversion, 2012). The staff round yearly burnup to the nearest whole gram of ^{235}U burned but keep a tally of previous year's excess deficiencies due to that rounding. For example, from 1992-93 the reactor staff calculated a burnup of 0.1549 grams which was rounded down to 0. For the next year the calculated burnup was 0.3766 g, to which the excess 0.1549 grams was added to yield 0.5315 grams. That number is then rounded up to 1 gram of ^{235}U burned,

assigned to a random fuel element from the core, and reported to the US-NRC. The following year starts with a 0.4685 (1-0.5315) gram deficiency due to the previous year's rounding up.

In the burnup analysis, the fuel material is tracked according to each plate, which yields a finer detailed distribution of burnup. In order to compare to MSTR reporting, the burnup was summed over for each fuel element and tabulated. Table 3.9 lists the amount of burnup in each fuel element of Configuration 120 according to MSTR reports and according to the burnup analysis.

Table 3.9. MSTR and simulation ^{235}U depletion distribution.

Fuel Element Type	Number	MSTR Reported Depletion (g)	Simulated Depletion (g)
Full Element	1	1	1.132
Full Element	2	1	1.058
Full Element	3	1	1.095
Full Element	4	1	0.892
Full Element	5	1	0.764
Full Element	6	0	0.783
Full Element	7	1	0.865
Full Element	8	1	0.782
Full Element	9	1	0.883
Full Element	10	1	0.681
Full Element	11	0	0.000
Full Element	12	0	0.892
Full Element	13	1	0.608
Full Element	14	1	0.598
Full Element	15	0	0.598
Full Element	16	0	0.000
Full Element	17	0	0.056
Full Element	18	0	0.000
Half Element	1	0	0.032
Control Rod	1	0	0.592
Control Rod	2	0	0.626
Control Rod	3	0	0.714
Control Rod	4	0	0.615
TOTAL BURNED (g)		11	14.226

According to the initial hot channel analysis performed, it might be expected that the control rod elements near the center of the core would have the most burnup. However, the control rod elements have only ten fuel plates; therefore there is less total fuel to burn. A more accurate comparison among fuel elements would be to consider the percent burnup in each element. However, for safety concerns the amount of ^{235}U in the core will not be released in this work. A more detailed understanding can be observed in the depletion map in Figure 3.4.

All of the fuel that has been in the core while the MSTR was at some power has burned some fuel so no fuel element should have zero burnup in Table 3.9. The MSTR's current method of assigning burnup to fuel elements does not give an accurate distribution. Table 3.9 shows that the simulation provides detailed burnup distribution among fuel elements.

Note the reactor staff rounds burnup to the nearest half of a gram. The total fuel burnup tallied in the simulation is more than the MSTR reported to the US-NRC by 3.476 grams. However, the MSTR staff reports annual values rounded to the nearest whole gram. This introduces significant error in the total burnup value. If the entire energy production is considered without the error introduced by cutting out value yearly, the burnup is listed at 14.5272 grams (MSTR, LEU Fuel Burnup Conversion, 2012). The simulated burnup analysis is 2.07% different than this value. The total burnup values are comparable but the distribution of material is essential to an accurate simulation of reactor characteristics.

3.4. POISON BUILDUP

In the studies done to obtain the optimal time step for the burnup simulation, significant poison buildup did not occur. However, twenty years of operation yielded more poison buildup. Significant isotopes (a high combination of concentration and cross-section) include ^{152}Sm, ^{147}Sm, ^{141}Pr, ^{143}Nd, and ^{99}Tc. These and other isotopes were incorporated into the updated MCNP model in order to calculate some core characteristics to check for improvement over the clean model.

4. BURNED MODEL RESULTS

With twenty years of fuel burnup and poison buildup incorporated, the new MCNP model (burned model) should be able to more accurately simulate experimentally measured characteristics than the clean model. Several reactor characteristics were simulated with both the clean and burned model and then compared to experimentally measured values. The burned model should more accurately simulate these characteristics and yield closer values to the experimentally measured values than the clean model.

4.1. CRITICAL ROD HEIGHTS

The heights of each rod are recorded in the MSTR power logbook when the reactor is brought to a critical state. There are multiple rod positions that will yield a critical MSTR because each rod has separate controls allowing for many combinations of critical rod heights. The rod heights for the reactor at a power of ten watts were obtained from the most recent appropriate recording. The low power of ten watts was chosen because operating at higher powers increases the temperature in the system which decreases the reactivity of the core requires the rods to be removed further. This change in temperature is not accounted for in the clean or burned MCNP models where the reactor is assumed to be isothermal. The reactor had been shut down the week previous to this day so there was no buildup of short-lived poisons. These critical rod heights are seen in Table 4.1 below and correspond to the experimentally measured k_{eff}=1 (critical).

Table 4.1. Critical rod heights

Control Rod	Height (in)
1	21.5
2	21.6
3	21.5
4	12.6

As a gage for model improvement, both the burned and the clean models were modified to have control rods at these positions and the effective multiplication factor was recorded. The results are presented in Table 4.2 below along with relative errors and represented graphically in Figure 4.1.

Table 4.2. k_{eff} values for burned and clean models at experimentally measured critical rod heights

Model	Keff	Relative Error
Experimental	1.00000	N/A
Clean	1.00399	0.00017
Burned	1.00250	0.00018

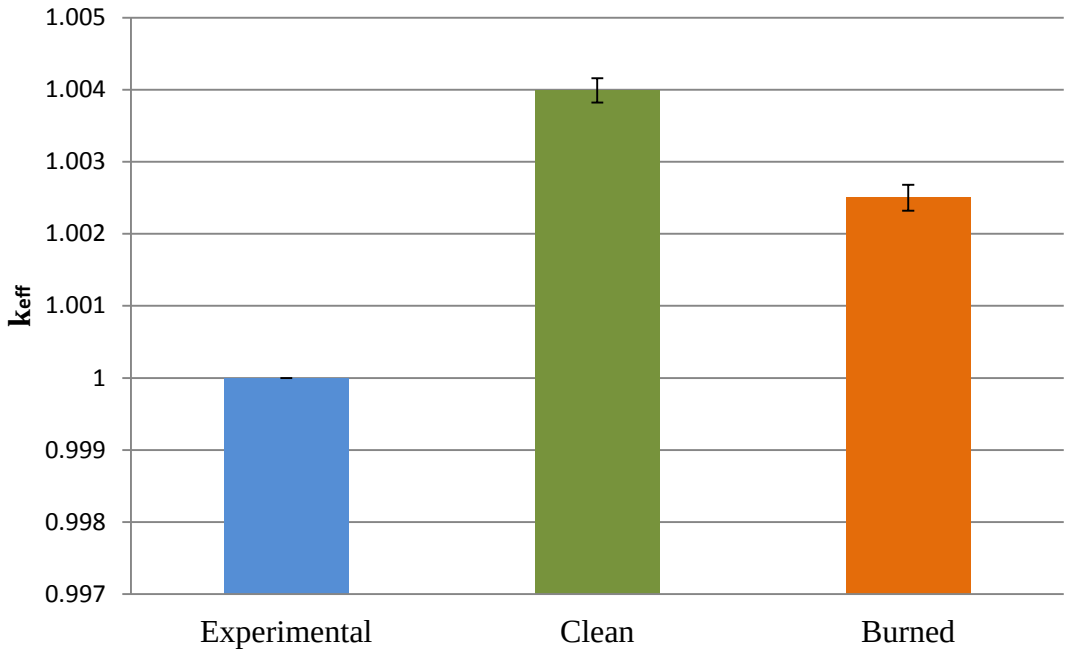


Figure 4.1. Comparison of k_{eff} values for burned and clean models at experimentally measured critical rod heights.

As expected, at critical rod height, the clean model has a higher k_{eff} value than the burned model because there is more fuel and less poison present, giving the core more positive reactivity. Both simulations yield values greater than 1, but with burnup material corrections, the burned model is 38% closer to the experimentally measured value.

4.2. EXCESS REACTIVITY AND SHUT DOWN MARGIN

Excess reactivity is the amount of reactivity in a reactor greater than that which is required to make it critical (European Nuclear Society, 2012). The US-NRC limits the amount of excess reactivity for every reactor in the United States. The excess reactivity of the MSTR with its current core configuration cannot exceed 1.5% Δk/k (Bonzer,

William, 2009). kcode is used to determine excess reactivity with simulations; the multiplication factor is obtained with all rods removed from the core. The excess reactivity then is found by equation 6.

$$\boldsymbol{\rho_{ex} = \frac{k_{wd}-1}{1}} \quad (6)$$

Where ρ_{ex} is the excess reactivity and k_{wd} is the simulated multiplication factor with all control rods removed. The excess reactivity of a burned core should be less than that of the clean core because there is less fuel and more poison present. The MSTR uses different methods to experimentally determine excess reactivity and records values for each configuration. Excess reactivity values from burned simulations, clean core simulations, and experimental measurements are summarized in Table 4.3 and shown in Figure 4.2.

From the US-NRC website, shutdown margin is defined as "the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth that is assumed to be fully withdrawn (European Nuclear Society, 2012)]." In addition to the rod with the highest rod being stuck, the regulating rod (control rod 4) is also not inserted for calculations of the MSTR shutdown because it is not a scrammable rod. Shutdown margin can be calculated using equation 7.

$$\boldsymbol{SDM = \frac{1-k_{ins}}{1}} \quad (7)$$

Where SDM is shutdown margin and k_{ins} is the effective multiplication factor with all the rods inserted except the rod of highest worth and the regulating rod. The US-NRC states that the MSTR must have a minimum SDM of 1% Δk/k (Bonzer, William, 2009). The SDM for Core Configuration 120 was calculated experimentally, simulated with a clean core, and simulated with a burned core. The results and errors are listed in Table 4.3 and shown in Figure 4.3.

Table 4.3. Excess reactivity and shutdown margin comparison between experimental, clean, and burned calculations for MSTR Core Configuration 120

	Exp (%Δk/k)	Clean (%Δk/k)	Clean Error	Burned (%Δk/k)	Burned Error
ρex	0.652	0.792	1.4256E-04	0.701	1.1917E-04
SDM	4.354	3.56	6.4080E-04	3.99	7.1820E-04

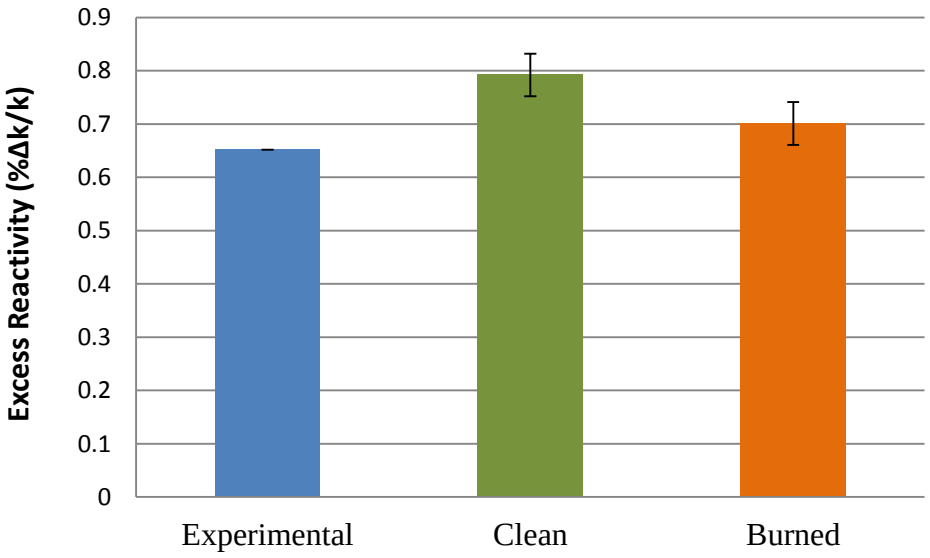


Figure 4.2. Excess reactivity comparison between experimental, clean, and burned calculations for MSTR Core Configuration 120

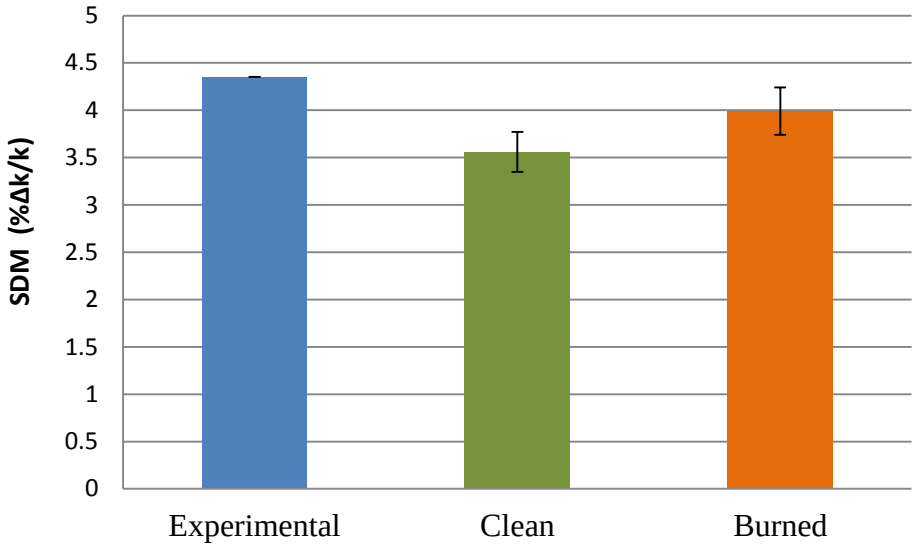


Figure 4.3. Shutdown margin comparison between experimental, clean and burned calculations for MSTR Core Configuration 120

The results show that the burned core simulation is closer to the values obtained experimentally than the clean core for both the SDM and excess reactivity. For excess reactivity the burned value is 35% closer and for SDM, it is 46% closer to the experimentally measured value. The excess reactivity is higher for the clean core because it has more fuel and less poison present. The incorporation significantly improved the MCNP simulation for these two core characteristics.

4.3. RELATIVE AXIAL FLUX PROFILES

The axial flux profile of the reactor cannot be measured directly so an experiment was developed to approximate it. A copper wire of known dimensions is placed inside one of the fuel elements and the reactor is brought to 500 W, operated for 10 minutes and then shut down. The wire is removed and allowed to cool in the pool such that the shorter

lived copper-64 isotope has decayed away and the wire will not be too active to handle. Then the wire is cut into 1 inch long segments, the segments are massed, and its activity is measured. The activity measured for each segment is divided by its mass to obtain a mass averaged activity as it is not possible to ensure the segment is exactly one inch. A significant amount of time passed between the measurement of the first and last segments so it is necessary to correct the activity value for decay. Because the activity of each segment is linearly related to the flux in the core, it is assumed that a graph of this data will have the same shape as a graph of the flux profile.

Two wires were irradiated and measured this way. The wires were placed in two different fuel elements (fourteen and two) but were irradiated at the same time. With limited vision of the fuel element, the only channel that the wire will fit in with certainty is the channel between the ninth and tenth plate because it has the lifting bale which acts as a guide for the wire directly above it. Ideally one wire would be located in the hottest channel and another in a channel with close to average predicted activity, but the element predicted to have the highest activity is a control rod element and does not allow for access to the copper wire. The two elements were chosen because of their contrasting positions in the core. Fuel element fourteen is near the center while two is on the periphery of Configuration 120. The locations can be seen in Figure 4.4.

	1	2	3	4	5	6	7	8	9
A									
B						S			
C					CR 4	FE 5	FE 1	FE 17	
D				FE 4	FE 8	FE 14	CR 1	FE 10	FE 2
E				FE 9	CR 3	FE 12	CR 2	FE 7	FE 3
F				CRT	FE 15	HC	FE 13	BRT	FE 6

Figure 4.4. Axial flux fuel element locations in Core Configuration 120

Once a profile was obtained for these wires, the activity was normalized to the maximum value. Both experimentally measured axial profiles are plotted in Figure 4.5. Simulated values using the burned core model can be seen in Figure 4.6.

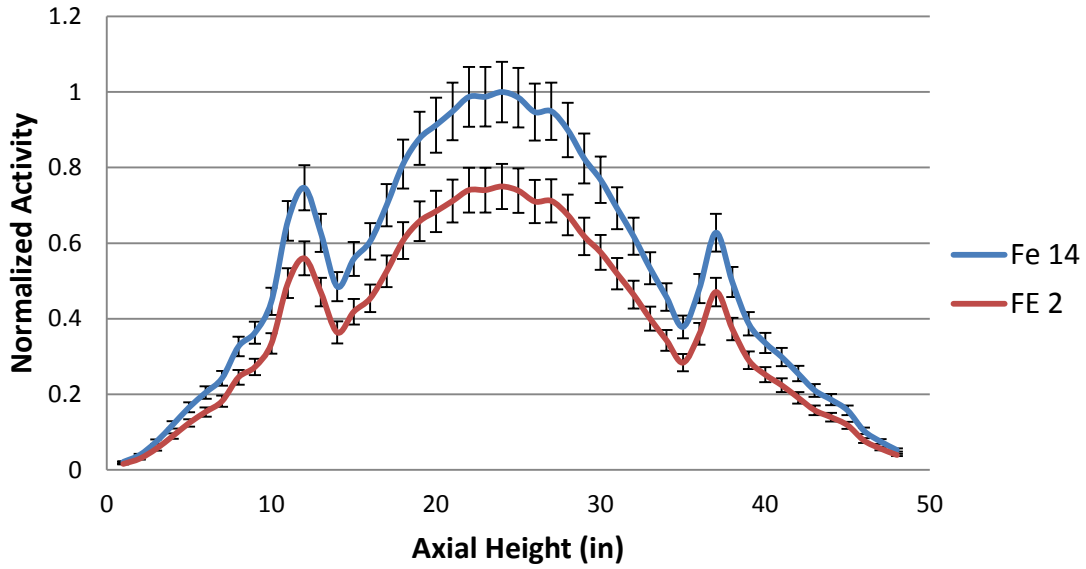


Figure 4.5. Experimentally measured axial flux profile, fuel elements fourteen and two

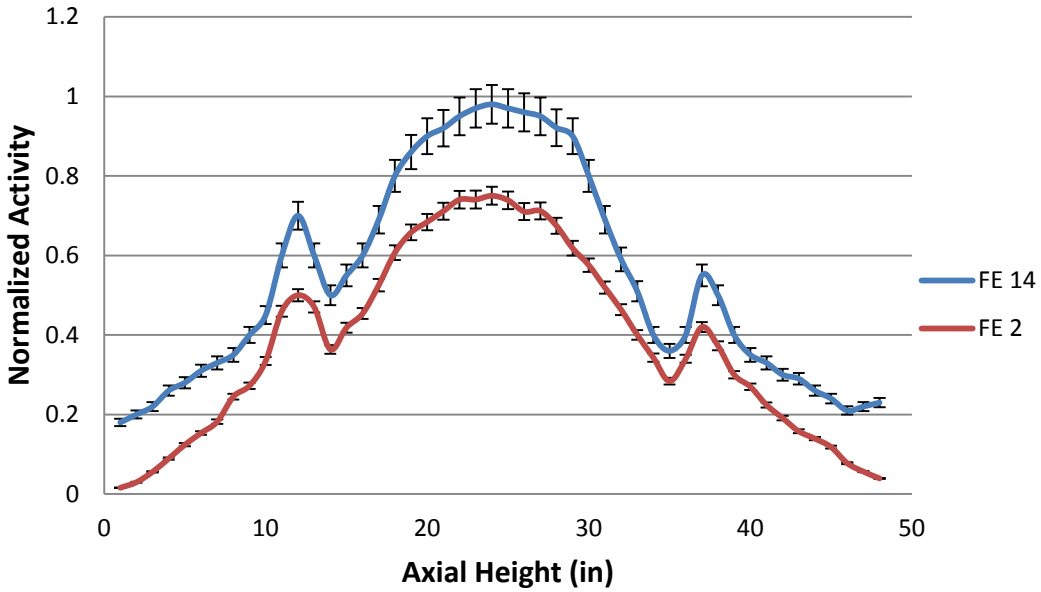


Figure 4.6. Simulated axial flux profile, fuel elements fourteen and two

The goal of this analysis is to verify the model's "radial" flux of the MSTR core and verify relative contributions from fuel elements to heat production. The ratio of the area under the experimental profile for element fourteen and element two compared to the simulated ratio and yielded an error of 8.54% ± 0.06%. Error is introduced experimentally by timing for decay, massing segments and measuring activity.

4.4. HOT CHANNEL DETERMINATION WITH BURNED MATERIAL

Using the burned core in simulations has yielded improved results over simulating with a clean core in calculating several core characteristics. By incorporating the burned material in the MCNP model and tallying energy deposition, an updated hot channel was obtained. Figure 4.7 is a map of the energy deposition in Core Configuration 120.

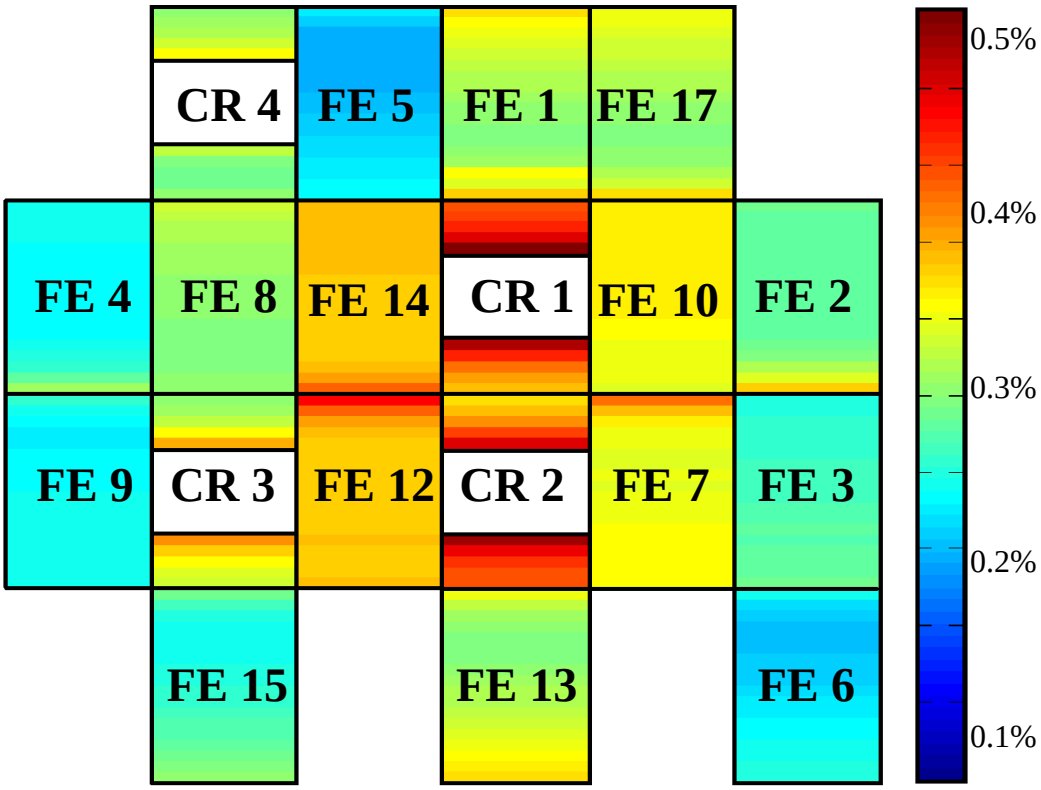


Figure 4.7. Fractional energy deposition in plates with updated fuel material

The hottest channel is still located between the sixth and seventh fuel plates of control rod one, despite having significant burnup. Incorporating the burned material did not shift the hot channel. However, the hot channel factor (ratio of maximum to average value of energy deposition) is 1.71, which is less than the hot channel factor of the clean core (1.85). This is expected because over the twenty years of MSTR operations the areas with higher neutron utilization lost more fuel, decreasing peaking.

4.5. ERROR ANALYSIS

When tallying energy deposition, flux, or determining $k_{eff,}$ MCNP outputs relative error with the tallies and kcode. These relative errors were multiplied by corresponding values and included as error bars as seen on appropriate graphs. When in operation for experimental measurements, the reactor was assumed to be critical (k_{eff}=1) with negligible error. For experimentally measured values, such as axial flux, error was propagated through the calculation and analysis of the experiment as follows (Knoll, 2000).

The standard deviation of a specific count, σ_{C_i}, was calculated using equation 8, where C_i is the value of that count.

$$\sigma_{C_i} = \sqrt{C_i} \tag{8}$$

Equation 9 gives the standard deviation of counts.

$$\sigma_C{}^2 = \frac{1}{\sum_{i=1}^{n} \frac{1}{\sigma_{C_i}{}^2}} = \frac{1}{\sum_{i=1}^{n} \frac{1}{C_i}} \tag{9}$$

Average background radiation is given by the equation 10, where b_i is one measurement of the background and n is the number of background measurements. Then the standard deviation for the background is given by equation 11.

$$b = \frac{b_1 + b_2 + \cdots + b_n}{n} \tag{10}$$

$$\sigma_b = \sqrt{\frac{b}{n}} \tag{11}$$

The standard deviation for the massing of each individual copper wire segment is assumed to be the accuracy of the scale, which is 0.0005 g. The standard deviation for the time of measurement is assumed to be 1 minute. The decay constant for Cu-66, λ, is 0.00091 min^{-1}. The activity of each segment is then given in decays/min/g by equation 12, where C_i is the measured activity of the i^{th} segment, b is the average background, λ is the decay constant, t is the time after removal from the core that the segment's activity was measured, and m_i is the mass of the i^{th} segment. $A_{0_i} = \frac{(C_i - b)e^{\lambda t}}{m_i}$ $A_{0_i} = \frac{(C_i - b)e^{\lambda t}}{m_i}$

$A_{0_i} = \frac{(C_i - b)e^{\lambda t}}{m_i}$

$$A_{0_i} = \frac{(C_i - b)e^{\lambda t}}{m_i} \tag{12}$$

The standard deviation of the activity of each segment is then given by equation 13 (Knoll, 2000).

$$\sigma_{A_{0_i}}^{\ 2} = \left(\frac{\partial A_{0_i}}{\partial C_i}\right)^2 \sigma_C^{\ 2} + \left(\frac{\partial A_{0_i}}{\partial b}\right)^2 \sigma_b^{\ 2} + \left(\frac{\partial A_{0_i}}{\partial t}\right)^2 \sigma_t^{\ 2} + \left(\frac{\partial A_{0_i}}{\partial m_i}\right)^2 \sigma_m^{\ 2} \tag{13}$$

Plugging into the formula then gives equation 14 for the error in activity of the copper wire segments.

$$\sigma_{A_{0_i}}^{\ 2} = \left(\frac{e^{0.00091t}}{m_i}\right)^2 \frac{1}{\sum_{i=1}^{n} \frac{1}{C_i}} + \left(-\frac{e^{0.00091t}}{m_i}\right)^2 \frac{b}{n} + \left(\frac{\lambda(C_i - b)e^{0.00091t}}{m_i}\right)^2 + \left(-\frac{(C_i - b)e^{0.00091t}}{m_i^{\ 2}}\right)^2 \sigma_m^{\ 2} \tag{14}$$

5. CONCLUSION

A burnup map of the current core configuration was developed to show the ^{235}U depletion distribution among the fuel plates (14.226 g total). With the burnup incorporated a new hot channel was determined to be located between plates six and seven of the fuel element of control rod three. One goal of this work was to develop an improved MCNP model of the MSTR. Incorporating the burned fuel material improved the simulation of core characteristics as follows;

- 38% closer in predicting multiplication factor at experimentally measured critical rod heights
- 35% closer in calculating shutdown margin
- 46% closer in calculating excess reactivity
- Burned model simulated relative axial flux measurements yielded a maximum difference of 8.54% from the experimentally measured value

The MSTR core was assumed to be isothermal for this analysis. Future work could include a thermohydraulics model to input accurate temperature profiles for more accurate nuclear data. Increasing the temperature of materials in the reactor system would decrease the value of k_{eff} because the MSTR has a negative temperature coefficient. A decreased value of k_{eff} would, in turn, improve the accuracy of all core characteristics simulated. Thus, incorporation of a thermohydraulics model should improve the neutronics model and more accurately simulate the MSTR system. Another source of error in these simulations is the fact that burnup analysis averages burnup over the entire the entire fuel plate. This may introduce error as the fuel in the center of the plate would have burned more than at the periphery of the plate because a higher neutron

flux exists at the center of the core. The burnup simulation utilized equivalent averaged powers and assumed the core was always in the ‘W’ mode to save computer time. Increasing time steps and more accurately modeling the ‘W’ versus ‘T’ positions of the core would more accurately simulate the burnup of the MSTR.

APPENDIX A

EXAMPLE MSTR POWER LOGBOOK

Table A.1. Example MSTR power logbook (Reisner, MSTR Power Logbook, 2012)

10/22/1992						
Time	1313	1325	1333	1340	1351	1357
Nominal Power (W)	20	2000	20000	40000	100000	0
Shim Rod 1 (in)	20.9	21	21.3	21.5	21.9	
Shim Rod 2 (in)	20.9	21	21.3	21.5	21.9	
Shim Rod 3(in)	20.9	21	21.3	21.5	21.9	
Regulating Rod (in)	13.1	14.8	14.4	14	14.1	
10/27/1992						
Time	1434	1459	1512	1521	1525	
Nominal Power (W)	10	100000	120000	80000	0	
Shim Rod 1 (in)	20.8	21.8	21.8	21.6		
Shim Rod 2 (in)	20.8	21.8	21.8	21.6		
Shim Rod 3 (in)	20.8	21.8	21.8	21.6		
Regulating Rod (in)	13.4	14.3	15.1	15		
10/28/1992						
Time	936	947	958	1003	1004	
Nominal Power (W)	10	100	10	10	0	
Shim Rod 1 (in)	21	21.2	21.2	21.2		
Shim Rod 2 (in)	21	21.2	21.2	21.2		
Shim Rod 3 (in)	21	21.2	21.2	21.2		
Regulating Rod (in)	14	12.6	12	12.5		
10/30/1992						
Time	1433	1443				
Nominal Power (W)	10	0				
Shim Rod 1 (in)	21					
Shim Rod 2 (in)	21					
Shim Rod 3 (in)	21					
Regulating Rod (in)	14.1					

APPENDIX B

MCNPX TIER 2 FISSION PRODUCT LIST

Table B.1. MCNPX tier 2 fission product list (Pelowitz, 2008)

Tier 2
^{95}Mo
^{99}Tc
^{101}Ru ^{103}Ru ^{105}Ru
^{102}Pd ^{104}Pd ^{105}Pd ^{106}Pd ^{108}Pd ^{110}Pd
^{107}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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VITA

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He graduated magna cum laude from what is now the Missouri University of Science and Technology with a Bachelor's of Science degree in Nuclear Engineering in May, 2011. He enrolled in graduate school in at Missouri University of Science and Technology and was a recipient of the Chancellor's Fellowship and National Academy for Nuclear Training Fellowship. He will complete his Masters of Science in Nuclear Engineering in December of 2012.