THE PENNSYLVANIA STATE UNIVERSITY
SCHREYER HONORS COLLEGE

DEPARTMENT OF MECHANICAL AND NUCLEAR ENGINEERING

PRIMARY CALCULATION OF THE LINEAR HEAT RATE GENERATION OF A BWR PIN IN THE ATR B-11 POSITION

NATHAN CHRISTOPHER ANDREWS
SPRING 2010

A thesis
submitted in partial fulfillment
of the requirements
for a baccalaureate degree
in Nuclear Engineering
with honors in Nuclear Engineering

Reviewed and approved* by the following:

Kostadin Ivanov
Professor of Nuclear Engineering
Thesis Supervisor

Arthur Motta
Professor of Nuclear Engineering and Materials Science and Engineering
Honors Adviser

* Signatures are on file in the Schreyer Honors College.

ABSTRACT

The goal of this project was to determine the linear heat generation rate of a boiling water reactor fuel pin placed into the Advanced Test Reactor at Idaho National Laboratory inside of a boiling water test loop. The loop was designed to be places in the B-11 position of the reactor. This required coupling the output of the computer codes CASMO-3 and MCNP5. First, a simulated depletion using CASMO-3 was performed to 25 GWD/MTU. The output from this was used in MCNP5 to determine the linear heat generation rate from the fuel pin. The flux in the fuel pin in the boiling water test loop was determined by comparison with a benchmark case. This flux was applied to the fuel pin in the test loop, to return the heat generation rate. The MCNP runs returned a maximum peak linear heat rate of 13.57 kW/ft for a gas gap width of 1.651 mm and a peak linear heat rate of 11.89 kW/ft for a width of .5 mm. These values are lower than the desired value of 15 kW/ft. All tests showed the desired result when diluted with He-3, dropping the power below 6 kW/ft. This demonstrated that the desired ramp testing could be performed.

ACKNOWLEDGEMENTS

The work of this thesis is due in large part to the aid of my mentors and fellow students at The Pennsylvania State University and Idaho National Laboratory. At Penn State, this work was made possible due to the contributions of the Reactor Dynamics and Fuel Management Group, particularly Shadi Ghrayeb and Zainuddin Karriem. Much of the work on which this thesis is dependent was performed by myself and three other interns at Idaho National Laboratory in the summer of 2009, Daniel Walter, Kevin Lyon and Dannielle Perez.

TABLE OF CONTENTS

INTRODUCTION

The Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) was designed to perform nuclear materials and fuels experiments in a high thermal neutron flux environment. It generates both a fast and thermal neutron spectrum roughly an order of magnitude higher than a current industrial reactor. This allows for vastly shortened testing times. For this purpose, it is the most effective nuclear reactor in the world. Its sole aim is to test materials and fuels rather than to generate electricity. Currently, tests are mainly performed in pressurized water reactor (PWR) environments, but several are also performed in gas environments. Pressurized water tests are performed for the Department of Defense, the lab itself, the nuclear industry and several universities. For the first 30 years of operation, the reactor was used primarily for testing of components for naval nuclear reactors. In 2007, the ATR was designated a National Scientific User Facility, and began offering university-led teams access to the reactor on the basis of competitive proposal selection. Despite extensive PWR testing capabilities, the reactor does not possess the capability to run tests in a boiling-water reactor (BWR) environment. Recently fuel vendors have shown interest in testing its fuel rods in the ATR. This capability is desirable since a large fraction of the world's reactors are based upon boiling-water technologies, which present unique fuel and materials issues.

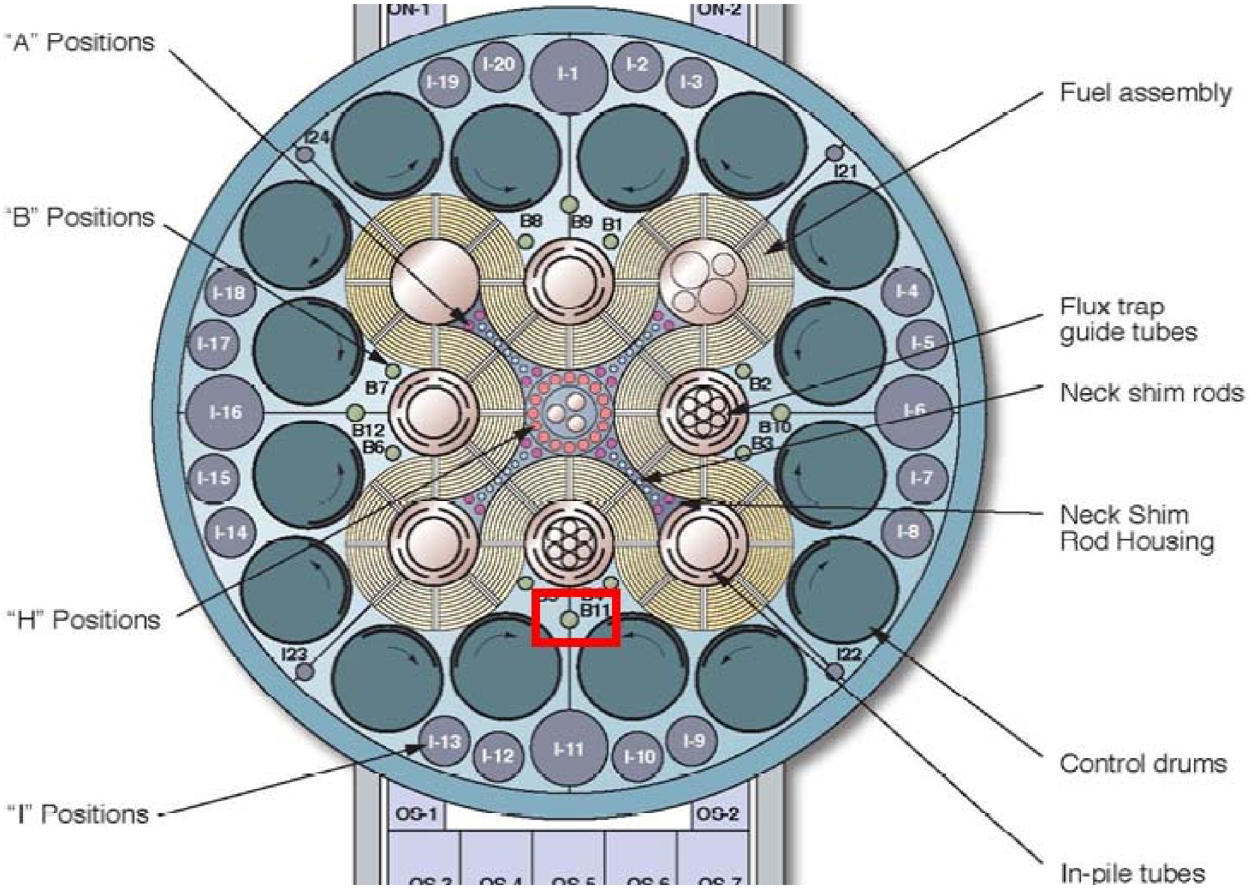

Figure 1: Axial Cross-Section of the Advanced Test Reactor at INL with B-11 Highlighted [5]

Experiments in the ATR are performed in various regions of the reactor core. The largest position available is roughly five inches in diameter, yet most are less than an inch and a half in diameter. For the B-11 position, which is 1.5 inches in diameter, a test loop for boiling water reactor fuels is being proposed. An axial cross-section of the reactor can be seen in Figure 1. One of the main tasks in the development of this endeavor is a neutronics and thermal hydraulic analysis. This analysis was performed under the Reactor Dynamics and Fuel Management Group at Penn State and its head Dr. Kostadin Ivanov.

To begin to address these issues, the ATR National Scientific User Facility is developing designs for a boiling water test loop through student intern projects. As part of this, I along with several other interns worked on creating an elementary design for the system. The summer internship project involved the design of the in-core portion of the loop as well as the basic secondary systems. Much more work needs to be accomplished, including design of an ASME-code pressure vessel insulated with a flowing gas-gap and separated from the primary coolant system, a test train design, provisions for ramp testing (stepped power increases) and neutronics calculations of a fueled experiment.

Before the development of the test loop can proceed, a feasibility assessment needed to be performed to ensure the appropriate linear heat generation rates could be generated with the neutron flux available in the B-11 position of the reactor core, which is where the loop will be inserted when ready. This assessment required neutronic modeling in both the fields of neutronics and thermal hydraulics. This required modeling using mainly CASMO-3 and MCNP5. These programs were available from the Reactor Dynamics and Fuel Management Group.

This honors project focused on conducting a neutronic assessment of a fueled test. The assessment assumed that the test piece is a segment of a BWR fuel rod 18" long with a burn-up of 25 GWd/MT. The analysis required the calculation of the isotopic composition of the burned fuel followed by analysis of the linear heat generation rate (power) of the fuel for the B-11 position with the SE and SW reactor lobes operating at a power of 23 MW. The goal for the test rig is to produce a linear heat generation rate of 15 kW/ft in the BWR fuel segment. If 15 kW/ft BWR fuel segment power cannot be attained at SE/SW lobe powers of 23 MW, the lobe power required to attain approximately 15 kW/ft should be determined.

The following tasks were specified as the ultimate goals of the research by Dr. Mitchell Meyer at Idaho National Laboratory, who serves as the ATR National Scientific User's Facility's Deputy Scientific Director.

1. Calculate or otherwise obtain the isotopic composition of a typical BWR fuel rod enriched to 5 wt.% U-235 and burned to 25 GWd/MT.
2. Determine the linear heat generation rate of the burned BWR fuel segment in the ATR B-11 position. Initially assume that the SE and SW reactor lobes are operating at a power of 23 MW.
3. The goal for the test rig is to produce a peak linear heat generation rate of 15 kW/ft in the BWR fuel segment. If 15 kW/ft BWR fuel segment power cannot be attained at SE/SW lobe powers of 23 MW, the lobe power required to attain approximately 15 kW/ft should be determined.
4. If time allows, investigate the affect of a ^{3}He blanket filling a gas gap around the fuel in reducing the power within the BWR fuel segment when operating at 15 kW/ft. Start with a 0.5 mm gap. Determine the width of the gap necessary to reduce the fuel segment power to 6 kW/ft. Dilute the ^{3}He blanket gas with ^{4}He if necessary to maintain a gap width near 0.5 mm.

The burn-up generation was performed using the CASMO-3 code and the neutronics analyses were all performed on MCNP5.

BACKGROUND: ATR/LOOP SPECIFICATIONS AND PREVIOUS WORK

In this experiment, the following basic requirements are specified. Water must enter and exit the top. The loop must be useable for multiple tests. The loop must be able to accommodate 18 in. long fuel pins which are 3/8 in. or ½ in. in diameter. There must also be instrumentation in the loop to monitor the fuel and flow conditions. More specification for the design of the loop are specified in the remainder of this section. These specifications need to be summarized as they are used to develop the inputs for both the CASMO and MCNP decks for this report. This section also includes information on previous work which was used as the basis for this project.

Operating Conditions:

To mirror the burn up of BWR fuel in an actual reactor, the following conditions need to be maintained. The necessary inlet and outlet conditions of the test loop are specified below, in Table 1 [4]. The operating conditions of the ATR are summarized in Table 2 [5].

Table 1: BWR Operating Conditions

Inlet Flow Conditions	1040 psia, 532 F
Outlet Flow Conditions	1040 psia, 547 F, $x = 14.6\%$
Design Conditions	1265 psia, 575 F

Table 2: ATR Core Specifications

Coolant Design Pressure	2.7 MPa (390 psig)
Coolant Design Temperature	115 C (240 F)
Max. Coolant Flow Rate	3.09 m^3/s (49,000 gpm)
Coolant Temperature (Operating)	52 C (125 F) inlet, 71 C (160 F) outlet
Reactor Thermal Power	110 MW_{th} for this test

Dimensions:

The diameter of the large B-position is 1.5 in. The test loop may extend beyond the top of the 48.0 inch ATR core. It cannot extend below the bottom of the core. In this study the 18 inches in the center, where the fuel pin will be located is the section of the loop of the most interest. The

Thermal and Fluid Characteristics:

The 18 in. fuel pin will have a heat generation rate ranging from 5 kW/ft (avg.) to 15 kW/ft (max.). It must undergo boiling in the portion of the loop where the fuel is located. Since BWR water is very close to pure water, the chemistry will not be very complex. There will be hydrogen injection of 200 ppm and tiny amounts of noble metal additions. Such small amounts are inconsequential from a neutronics point of view and can be ignored.

Neutronics:

The peak fluxes at 62 cm. in the core are roughly 1.1×10^{14} (2200 m/s) and 1.6×10^{13} (above 1 MeV). An in-depth flux profile with 500 energy groups for the entire core was used for all of the inputs; however it cannot be distributed due to export controls. The reactor is used for Department of Defense testing. The normalized centerline spectrum for the B-11 position of the ATR can be seen in Figure 2. For axial locations 15 inches above and below the centerline, the breakdown of the flux spectrum is the same. The total flux in relation to average core flux as a function of various axial locations in the core can be seen in Figure 3. The burn-up of the fuel will be 25 GWd/MTU burnup (+/- 5). [5]

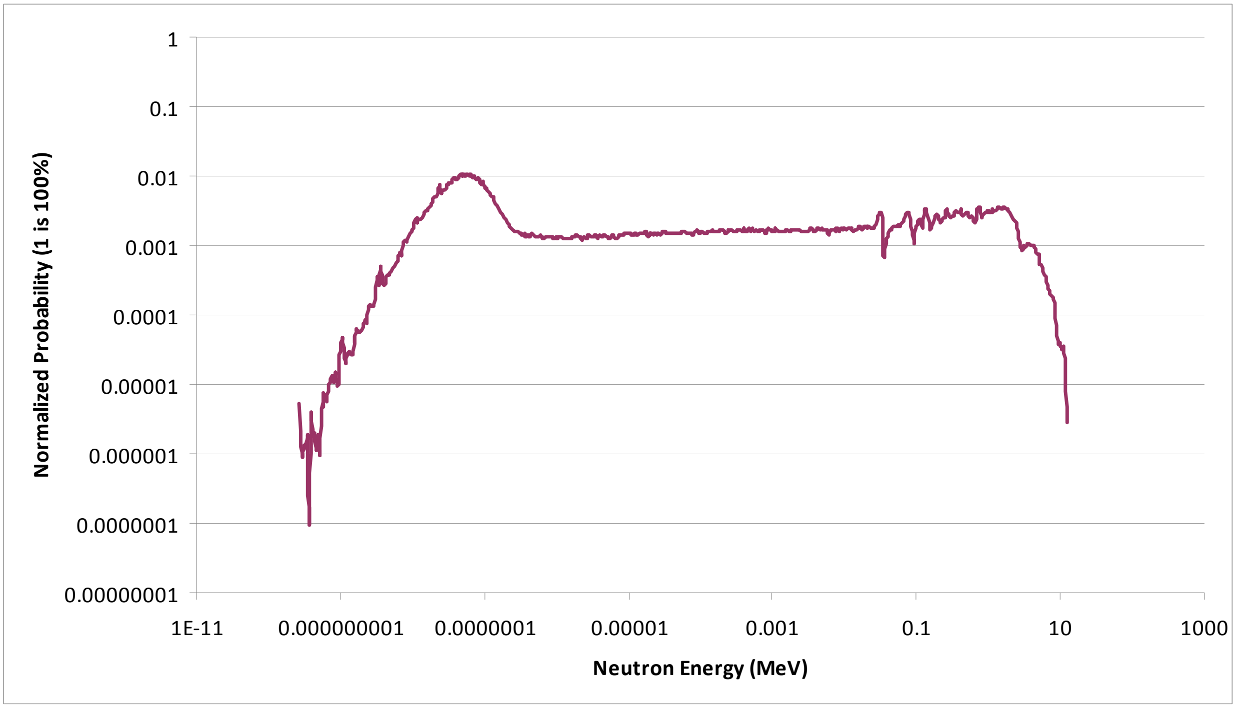

Figure 2: ATR B-11 Normalized Flux Spectrum at the Centerline

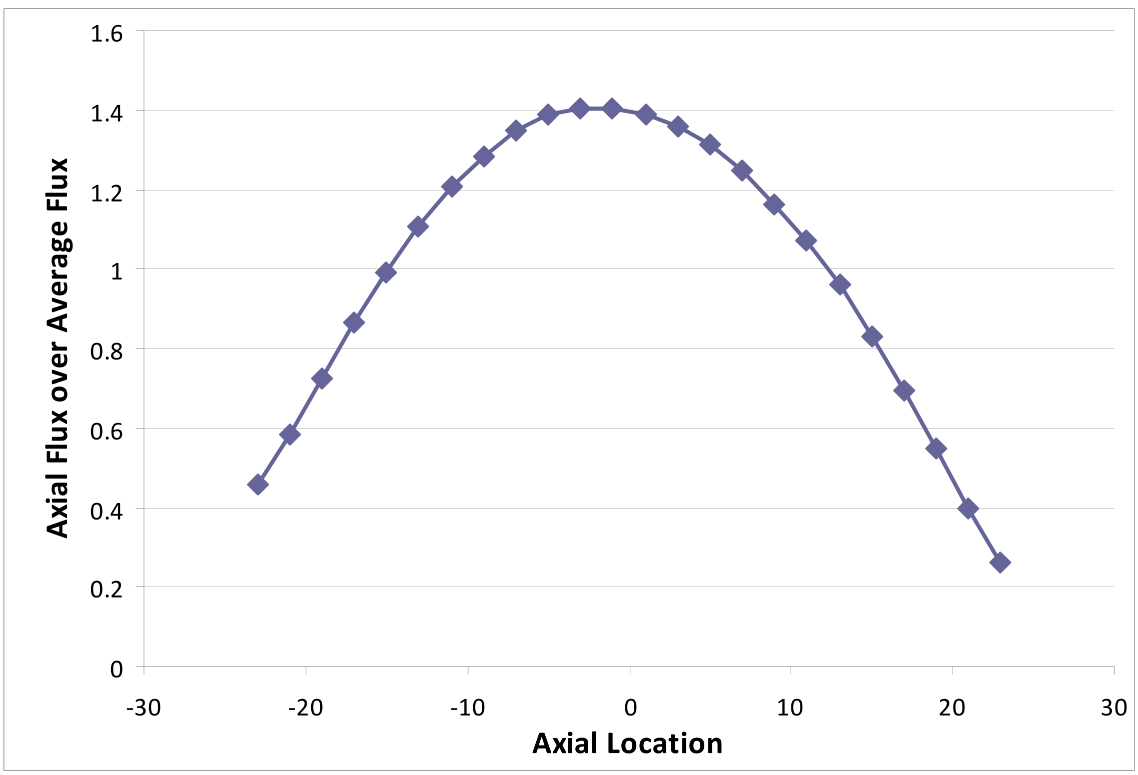

Figure 3: Axial Flux over Average Flux as a Function of Axial Location (B-11 Position)

Power Ramping and Test Duration:

Each test will be in the core for two 56 day cycles. 106 days will be at steady state operation conditions and 6 days will be spent during power ramping. The power ramping will take the pin to 15 kW/ft and the steady state operation will occur at 6 kW/ft.

Summary of Past Relevant Work

Calculations were performed to determine the necessary thickness of the gas gap and the required heat for boiling. Important flow characteristics were also determined, including flow rate and Reynolds number. Current wall thicknesses are based upon pressure ASME-code analysis and current gap distances will accommodate all thermal expansion. A basic design of the test-train was created; however, it is not currently designed to be removed at the end of cycles. The axial cross section of the test loop can be seen in Figure 4. It should be noted that the current design is based on SS 316L and not SS 348. However, all of the dimensions, with the exceptions of the BWR fuel pin diameter and the gas gap, in the figure are the same as those in the test loop developed in the summer of 2009. The dimensions of gap were also altered to maintain thermal specifications. The composition of the gas gap was also altered in this study for the same reasons. These dimensions were used for the neutronic modeling of the loop.

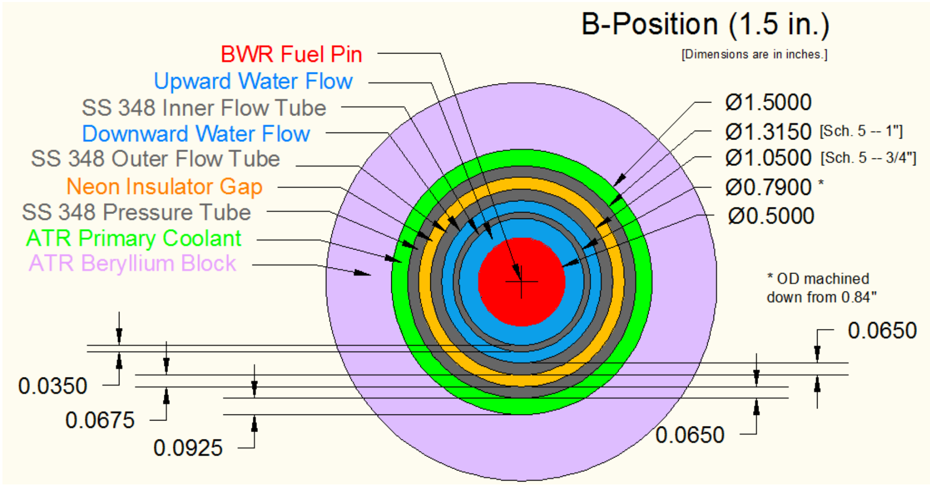

Figure 4: Axial Cross Section of the Boiling Water Test Loop from Summer 2009

The basic layout of the test train can be seen in Figure 5.

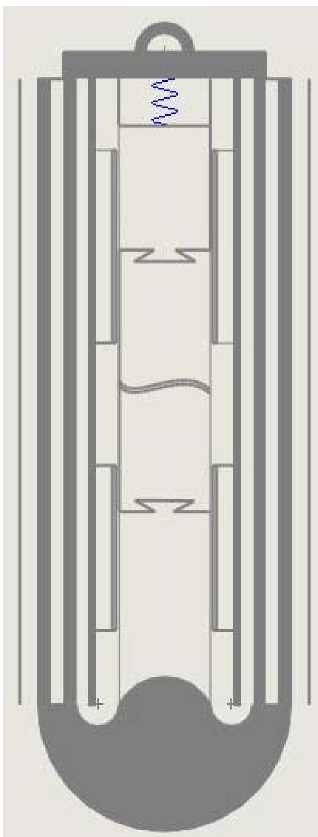

Figure 5: Basic Test Train Design from Summer 2009 (Not to Scale)

METHOD

In order to model the pin's performance in the ATR reactor, two separate codes were used. The first was CASMO-3, which simulates burn-up in a light water reactor environment. It returns the isotopic composition of a pin cell after a given depletion, given an initial composition. The output from this was then used as an input for the pin cell portion of the loop in an MCNP5 input deck. MCNP5 is usually used to determine the criticality of nuclear reactors. However, the code also possesses the capability to determine the amount of heat generated in a certain portion of the input. The CASMO input and output were only the BWR pin cell, which will be inserted into the loop. On the other hand, the MNCP input deck had to take into account the entire loop, as the reactivity and the number of neutrons available for use is a dependent on the structure of the loop and the materials that surround the fuel pin in the loop, not just the pin itself. The modeling programs used are summarized in Table 3, the task numbers correspond to the tasks listed in the introduction.

Table 3: Task and Method for Completion

Task Number	**Method for Completion**
1. Composition	CASMO-3 modeling
2. Linear Heat Generation	MCNP5 modeling
3. Other Core Configurations	MCNP5 modeling
4. Helium Blanket Analysis	MCNP5 modeling

The CASMO model was done for steady state operation of a BWR pin cell to 25 GWD/MTU. The isotopic composition generated in the 25 GWD/MTU section of the out was then used as the input for the pin cell section of the boiling water test loop. All of the isotopes generated in the burn-up were included except the saturated and unsaturated fission products. The dimensions for the boiling water test loop were based on previous work done by myself and other INL interns in the summer of 2009. The MCNP5 calculations were performed in neutron-photon mode to model both neutrons and photons, since a significant amount of heat is generated by gamma rays in addition to fission energy. The source neutrons, in this model, were treated as an external source. They were set to originate from a plane bisected by the x-axis of the sample. They were broken down into probability bins, using the flux profile show in Figure 2 and set to proceed only in the positive x-direction towards the test loop. This ensured that all of the neutrons hit the test loop. Leakage in the positive and negative z-directions could be neglected because the normalized energy profile of the flux spectrum is homogenous in the axial direction. The loop is symmetric and the boundary conditions are all reflective which makes a neutron source originating from a plane viable for this model.

In order to benchmark and scale the flux used in this study, a separate MCNP input had to be developed based upon a beryllium reflector in the B-11 position. This is needed because the flux values are dependent upon material composition and orientation of the B-11 position. Therefore a known geometry, with a known flux is used to find the flux present in a different geometry. The composition and dimensions of this filler are based upon the values taken from the sample MCNP model of the ATR core found in the International Handbook of Evaluated Criticality Safety Benchmark Experiments.

Tally six was used to find the amount of heat generated in the fuel pin portion of the loop per incident neutron. It determines heating rate as a function of mass and the number of source particles. This was then used to determine the linear heat generation rate of the fuel pin. The

MCNP5 output was in the units of MeV/g per incident neutron. This was normalized with the incoming current, giving the overall linear heat generation rate. The results of the code were then checked against the required operation specifications of the code.

The width and composition of the gas gap was varied to ensure the loop maintained the correct linear heat generations rate. The appropriate composition of the gas gap and its diameter was determined in the MCNP5 portion of the modeling. This was done through running the code multiple times with different dimensions and compositions. In comparison, the CASMO input deck only had to be run once. In order to ensure accuracy of the numbers generated, the MCNP5 code was run to 2,000,000 neutron generations before termination, to ensure the entire flux spectrum is taken into account. All of these runs used flux values which were altered by the scaling factor developed through comparison of the values from the runs with the benchmark data.

THEORY

CASMO uses a user-specified to determine the isotopic burn-up of nuclear reactor fuel pins and assemblies. It uses the isotopic abundances specified by the user, as well as user-specified reactor operating conditions to return a prediction of the isotopic abundance at different fuel burn-up steps. It does this through utilizing multi-energy group cross sectional data. It generates a neutron spectrum for the given operating conditions and then applied this spectrum to the energy group cross sectional data. This data is then used to return the abundances. [1]

MCNP is a Monte Carlo transport code that can be run in several different transport modes with neutrons, photons or electrons. Several of these modes can be run simultaneously. The code uses a continuous energy spectrum from 10^{-11} MeV to 20 MeV for all isotopes, and up to 150 MeV for certain isotopes. The photon energy regime runs from 1 keV to 100 GeV, and the electron regime goes from 1 keV to 1 GeV. Given a geometry and material specification, the code will calculate a number of different normalized factors including current, flux and energy generation. One of the main assets of the program is its ability to calculate the eigenvalue k_{eff} for a reactor, when this is done the source is 14 MeV, which represent newly released fission neutrons, at the origin of the coordinate system, unless otherwise specified. This is done through the KCODE command in the program. [2]

Monte Carlo methods for solving problems are different than deterministic methods. Deterministic methods use average values to solve the transport equation. Monte Carlo methods, on the other hand, solve the same problem by simulating the histories of individual particles and then recording different features (tallies) of their average behavior. The method is used to theoretically duplicate probabilistic processes. In the problem solved here, it is simulating neutron and photon interactions with matter. Its probabilistic equations are based upon a random number generator. (This is where the name ‘Monte Carlo’ originated.) This process is applied to every aspect of the transport equation for all particle modes involved. The process is repeated until the number of histories requested is reached. [2]

MODELING

CASMO-3 INPUT

As described in the modeling section, this project involved a CASMO-3 pin cell depletion coupled with a MCNP5 heat generation tally. The rationale behind the development of each of the input decks is summarized here.

BWR Pin Cell Depletion to 25 GWD/MTU

The first part of this thesis is to develop the isotopic abundance of a BWR pin cell, which has been depleted to 25 GWD/MTU. The modeling program which can best do this is CASMO. Using CASMO-3, the specified depletion was performed. As a result of the run, isotopic abundance of the fuel region was determined for both heavy nuclides and fission products. Both the CASMO-3 input deck and the resulting output are summarized in the following sections. [1]

Input Decks

In the first part of this model, a BWR pin cell was depleted to 25 GWD/MTU. This depletion was performed using CASMO-3. Data for the input deck was taken from Dr. Lawrence Hochreiter’s text entitled *Elements of Nuclear Reactor Design*, which serves as a textbook for the Reactor Thermal Hydraulics course at The Pennsylvania State University. The pin cell and environment characteristics are based on Table 2-16 (BWR Core Geometry and Thermal-hydraulic Characteristics BWR/6) on page 2-107. Note that all text to the right of a “*” symbol are comments. The input deck can be found in Appendix A, a card by card breakdown of the input is listed in the following paragraphs. [4]

The average fuel temperature (TFU) was based on Figure 5-24 (Peak Fuel Centerline, Average and Surface Temperatures during Fuel Rod Lifetime versus Linear Power) on page 5-42, which gives average fuel temperature as a function of linear heat generation rate. In this case, a core-averaged linear heat generation value of 6 kW/ft returned an average fuel temperature value of 1060 K. Average moderator temperature (TMO) and average void fraction values (VOI) were both taken from Table 2-16. Since this is a BWR pin cell, initial boron concentration (BOR) was set to zero. The fuel card (FUE) was set based upon a theoretical UO_2 density of 94%. The "1" at the beginning of the card refers to UO_2. The pellet was enriched to 5% U-235. Therefore, the weight percent U-235 was determined as follows. [4]

$$Enr\% \times \frac{M_{U-235}}{M_{UO2}} = 5 \times \frac{235}{235 + 16 \times 2} = 4.401W\%$$

The dimensions of the pin-cell (PIC) were once again based upon Table 2-16. Values, given in inches, were converted to centimeters. The first number in the card is the outer pellet radius. The second is the inner cladding radius. The third is the outer cladding radius. The fourth is the outer pin cell radius. Since the pin cell depletion functions in CASMO-3 use circular geometry, the pitch needed to be converted into an outer radius. This was done in the following method, where p is the pitch and r is the outer radius. [4]

$$\pi r^2 = p^2$$

The power density (PDE) and pressure (PRE) cards are in units of W/gU and bar respectively. The depletion card (DEP) was user specified to 25 GWD/MTU. Since the default cladding material is Zr-2 in CASMO-3, and this is a standard cladding for a BWR, no cladding material needs to be specified. [4]

CASMO-3 OUTPUT

This section summarizes the relevant output data from both the CASMO-3 code. The CASMO-3 output summarizes the isotopic abundance at 25 GWD/MTU. When the input deck is run through CASMO-3, it determines isotopic abundance at depletion steps of 5 GWd/MTU to 25 GWd/MTU. Before any depletion is performed by the program, it first calculates the initial isotopic abundance of all four regions in the pin cell. These values can be seen in Table 4.

Table 4: Nuclides in Pin Cell before Burn-up

Region	Nuclide	ID #	# Density (#/cm3)
Fuel	U-235	92235	1.00E+21
	U-234	92234	8.04E+18
	U-238	92238	2.14E+22
	O	8000	4.40E+22
Coolant	H	1001	2.98E+22
	O	8000	1.49E+22
	B	5000	0.00E+00
Air	O	8000	3.76E+19
Clad	Zr-2	302	4.30E+22

Given these initial values, the depletion was continued until 25 GWd/MTU was reached. When this occurred, the program ended and provided an output for isotopic abundance. Table 5 contains the isotopic abundance of heavy nuclides present at 25 GWd/MTU and Table 6 contains the isotopic abundance of fission products at 25 GWd/MTU.

Table 5: Heavy Nuclides in Fuel at 25 GWD/MTU

Nuclide	ID #	# Density (#/cm3)	Nuclide	ID #	# Density (#/cm3)
U-235	92235	5.23E+20	Am-241	95241	8.72E+17
U-236	92236	9.69E+19	Am-242m	95242	1.55E+16
U-237	93237	9.67E+18	Am-243	95243	5.98E+17
Pu-238	94238	2.53E+18	Cm-242	96242	1.75E+17
U-238	92238	2.10E+22	Cm-244	96244	1.23E+17
Pu-239	94239	1.95E+20	U-234	92234	5.28E+18
Pu-240	94240	3.59E+19	U-239	92239	1.08E+16
Pu-241	94241	2.67E+19	Np-239	93239	1.56E+18
Pu-242	94242	3.45E+18			

Table 6: Fission Products in Fuel at 25 GWD/MTU

Nuclide	ID #	# Density (#/cm3)	Nuclide	ID #	# Density (#/cm3)
Kr-83	36083	2.28E+18	Pm-148m	61248	6.20E+16
Rh-103	45103	1.65E+19	Sm-149	62149	2.19E+17
Rh-105	45105	3.72E+16	Sm-150	62150	7.75E+18
Ag-109	47109	1.72E+18	Sm-151	62151	8.33E+17
Xe-131	54131	1.35E+19	Sm-152	62152	2.73E+18
Cs-133	55133	3.41E+19	Eu-153	63153	2.82E+18
Cs-134	55134	3.24E+18	Eu-154	63154	8.35E+17
Xe-135	54135	1.27E+16	Eu-155	63155	4.68E+17
Cs-135	55135	2.11E+19	Sat. FP	401	7.74E+20
Nd-143	60143	2.81E+19	Unsat. FP	402	1.68E+20
Nd-145	60145	1.91E+19	I-135	53135	1.54E+16
Pm-147	61147	5.46E+18	Pm-149	61149	3.40E+16
Sm-147	62147	2.12E+18	Gd-155	64155	1.02E+16
Pm-148	61148	2.55E+16			

Since the isotopic abundance of the oxygen in the fuel, the air in the gas gap, the Zircaloy in the cladding and the coolant itself will not change; these values can be summarized based upon initial values. They are contained in Table 7.

Table 7: Other Nuclides in Pin Cell at 25 GWD/MTU

Region	**Nuclide**	**ID #**	**# Density (#/cm3)**
Fuel	O	8000	4.49E+22
Coolant	H	1001	2.98E+22
	O	8000	1.49E+22
	B	5000	0.00E+00
Air	O	8000	3.76E+19
Clad	Zr-2	302	4.30E+22

MCNP5: INPUT

This section summarizes the input used for the MCNP5 portion of this experiment. It includes the rationale for the all of the inputs used.

Surface and Cell Cards

Surface cards define the surfaces within the model. These surfaces are then used to define volumetric cells, where the collisions and irradiation will occur. In this model the first step was to define the surface cards. These were based on the dimensions of the boiling water test loop design from summer 2009. These dimensions can be found in Figure 4. The only dimensions which were altered were the width of the outer gas gap and the diameter of the outer pressure tube, which had to be increased as the width of the helium gas gap was increased. These cards are all cylindrical in nature. They are used to define all of the cylindrical portions of the model. Cells 1 to 10 account for the loop itself. Cells 1, 2 and 3 define the fuel pin. Cells 4 through 10 define the rest of the loop.

The total height of the model was 1 cm. This is contained in surfaces 5 and 6, which limit height. Surfaces 7, 8, 9 and 10 create the boundary of the model in the x and y directions. They are used to create a square void region outside of the loop; this is cell 11. This is important because the source is placed on surface 7, outside of the loop itself. Outside of these surfaces is a graveyard region where all particles have no importance; this is cell 12. With the acceptance of this graveyard, the importance of all particles is considered equally in all cells.

The materials used in the cell definition are specified later in the input deck in the material cards. The densities used in the cells are dependant upon the materials used. The stainless steel and fuel pin portion densities are dependant on the ASTM standards and data returned from CASMO-3. The innermost water portion, cell 4, uses the coolant density from CASMO-3. It therefore takes into account the void. Other water cells and the gas gap cell densities were defined using the given state points of densities and temperature. The gas gap density was based on a temperature of 140 F and 1040 psig. This is the pressure of the boiling water test loop (BWTL) and the temperature of the ATR coolant. The outer water flow section density was based on inlet BWR operating conditions of 1040 psia and 532 F.

Material Cards

The materials for the loop are defined isotopically by either weight percent or atom fraction in the material card section of the MCNP5 input. The materials cards also utilizes established the cross sectional data used for calculations. Material 1 is the fuel pin. The composition for it is based upon the output from CASMO-3. The number densities returned from the burn-up calculation at 25 GWD/MTU were normalized and used as the materials present in the pin, with the exception of the saturated and unsaturated fission products, since they are a compilation of the other isotopes present. The cross sectional library used was 72, which is defined at 900 K. This corresponded well to the operating temperature of the uranium. The fuel pin gas gap (material 2) and the cladding (material 3) are both also defined in the same manner, using the output data from CASMO-3. They are set to cross sectional decks of 900 K and 600 K respectively. Material 4 is the water immediately surrounding the fuel pin. It is defined using cross-sectional deck 53 and scattering deck 62, which are the closest ones to the operating conditions of the boiling portion of a BWR. Material 5 is stainless steel 316L. Its composition is based on the average allowable composition parameters. It is defined at 600 K using decks 42 and 71. Material 6 is water and is defined in the same manner as material 4. Material 7 is the helium gap. It contains He-3 and He-4. The deck used to describe it is 66, which is set to room temperature. Material 8 is the ATR primary coolant. It uses deck 42, which is set for 300 K, this is close to the operating conditions of the primary systems.

Source Definition

The neutron source was based upon the energy spectrum of the flux breakdown provided by INL. The source was broke down into bins at base 10 jumps from 1.0E-10 MeV to 14 MeV. These bins were based on normalized probability. They can be seen in Figure 6. The source was set along surface 7, which is the edge of the void graveyard and part of the inner void region. It was given equal probability to originate at any location in the y-direction or z-direction along this plane, which also had part of the loop. The particles themselves were set to head in the positive x-direction towards the loop. Since the loop is symmetric and homogenous in terms of source energy, there is not a need to account for other source directions. It also mirrors the direction of the neutrons travel in the core. The direction of the source can be seen in Figure 7. The KCODE command was not used since this command would use only 14 MeV neutrons and appropriately account for loss in the rest of the loop. Also, this is not a criticality calculation.

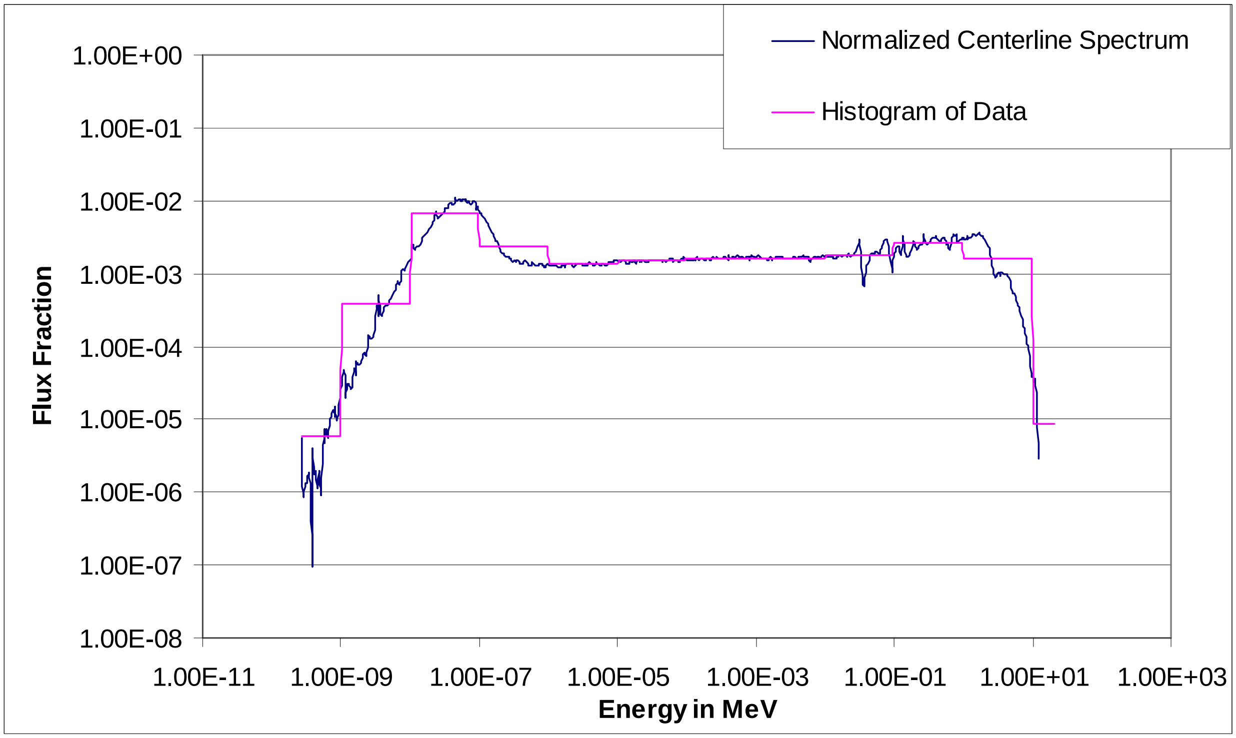

Figure 7: Energy Breakdown of Source Spectrum

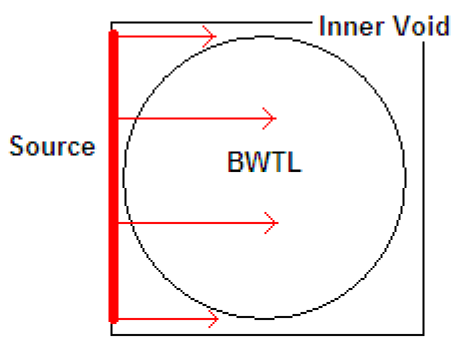

Figure 8: Location and Direction of the Source (Not to Scale)

Tally Cards

Tally 6 and tally 4 were the only tallies used in this model. Tally 6 returns the energy generated per gram per source particle: MeV/(g*particle). It includes both neuron and photon heating. This tally was performed for cells 1, 2 and 3, which comprise the fuel pin in the center of the BWTL.

The tally takes into account heating resultant from both neutrons and photons. Tally 4 was set to return the flux averaged over the cells 1, 2 and 3, which correspond to the fuel pin.

Other Data Cards

This section summarizes other cards in the input which are important, but not covered elsewhere. The calculation was run in MODE N P to account for both neutrons and photons. The calculation was run for 2,000,000 neutron histories to accurately model the source (NPS 2000000). The NONU card eliminates neutrons resultant from fission and treats fissions events as a loss to fission. This makes the assumption that the neutrons produced in the B-11 position are already accounted for in the SDEF source. The NONU card allows the SDEF card to remain the sole producer of neutrons in the B-11 position. It maintains the specified energy spectrum for the entire run. The PRINT 110 command returns the energies and locations of the first 50 neutrons.

Benchmark

The benchmark case was based upon the materials and surfaces used in the ATR criticality model, which is used to benchmark experiments in the International Handbook of Evaluated Criticality Safety Benchmark Experiments. The rest of the input deck was kept the same as the deck which was used for the BWTL, with the exception of Tally 6. It was neglected to speed-up run time because it returns heat generation information and only the flux information is of importance in the benchmark case. [3] This input for this benchmark run can be seen in Appendix B.

MCNP5: OUTPUT

The MCNP5 output summarizes the thermal characteristics of the BWR pin at various gas gap compositions and sizes. This section also summarizes the first fifty neutron histories of a sample run. These histories can be seen in Figure 9 and Figure 10.

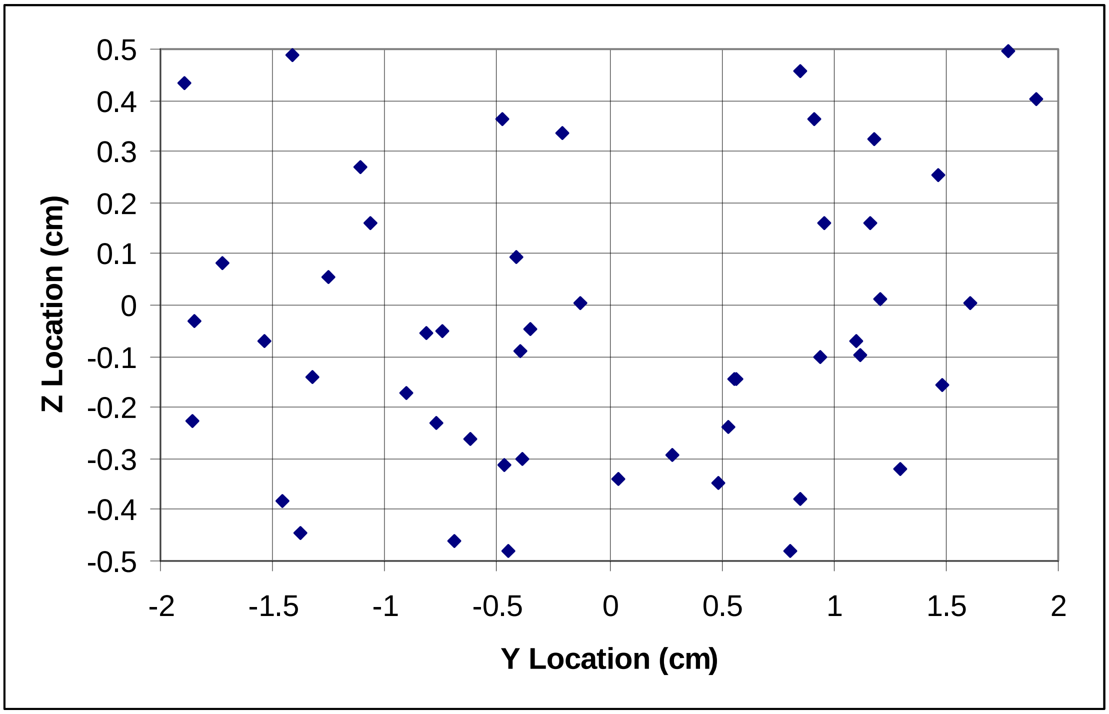

Figure 9: Y and Z-Locations of the Source Neutrons on the Source Plane at -2 cm X-Plane

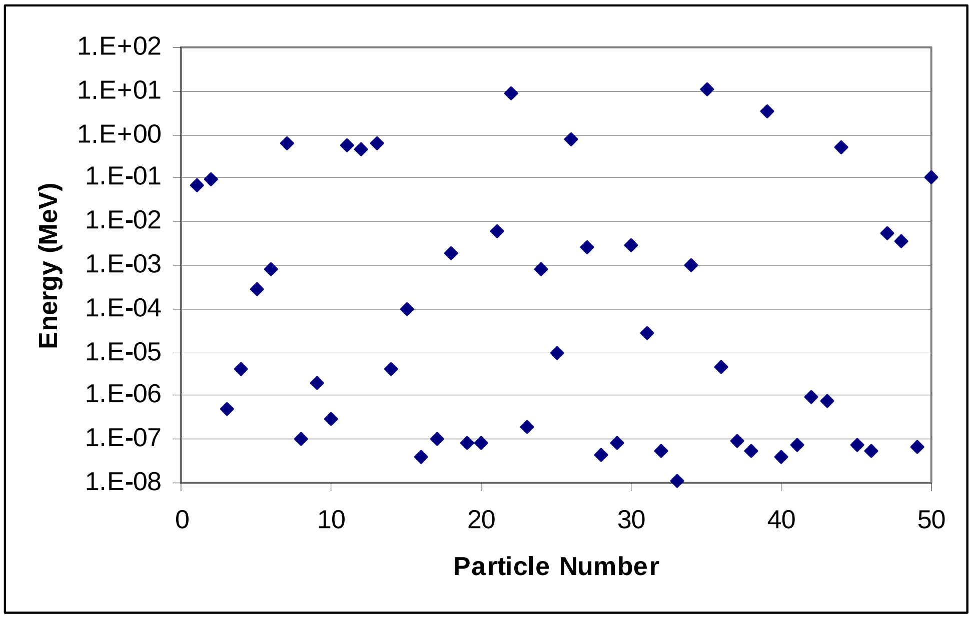

Figure 10: Energy of the First Fifty Neutrons on a Logarithmic Scale

The results of all the runs performed in this experiment are summarized in Table 8 (on the next page). The table also includes the values that were altered in the input deck to produce the result. These values are averaged over the entire fuel pin in the BWTL model. The table also contains the benchmark case, a case with neon as the filler gas and a case with the helium gap at half density. The table also includes the scaling factor which is a ratio of Tally 4 for the specific run to Tally 4 for the benchmark case. The scaling factor can be defined with the following equation.

$$F_{Run} = \frac{T_{4-Run}}{T_{4-Bench}}$$

Table 8: MCNP Output Data for Tally 4 and Tally 6

Run	He Gap O.D.	Outer Tube O.D.	Gap Width	Flux Avg. in Pin (T4)	Scaling Factor	He-3	Heat Generated (T6)
#	cm	cm	mm	1/cm2s	-	Fraction	MeV/g
1	1.3835	1.67005	0.5	1.21	0.235	0.0	1.28
2	1.3835	1.67005	0.5	1.13	0.221	0.5	0.998
3	1.3835	1.67005	0.5	1.07	0.210	1.0	0.789
4	1.4335	1.67005	1.0	1.25	0.243	0.0	1.41
5	1.4835	1.67005	1.5	1.29	0.253	0.0	1.56
6	1.50495	1.67005	0.1651	1.32	0.257	0.0	1.63
7	1.5335	1.7399	2.0	1.42	0.277	0.0	1.49
8	1.3835	1.5486	0.5	1.15	0.224	0.0	1.64
9	1.4335	1.67005	1.0	1.03	.201	1.0	0.583
10	1.4835	1.67005	1.5	0.999	0.195	1.0	0.481
11	1.50495	1.67005	0.1651	0.989	0.193	1.0	0.448
Neon	1.3835	1.67005	0.5	1.21	0.235	0.0	1.28
Half Density	1.3835	1.67005	0.5	1.21	0.235	0.0	1.28
Benchmark	-	-	-	5.12	-	-	-

*Note: Since the original units for the loop are in inches, the significant digits of the loop dimensions are off.

ANALYSIS

Using the heat generation from the output data, the linear heat generation rate was calculated, using equation 1.

$$\frac{\bar{\phi}_{pin} A_{XC} E_{gen} M_{pin}}{H_{pin}} = q' \qquad \text{Equation 1}$$

where: $\bar{\phi}_{pin}$ = Pin Average Neutron Flux
A = Cross Sectional Area of Pin
E = Energy Generated per gram per neutron
M = Mass of the Pin in the Model
H = Height of the Pin in the Model
q' = Linear Heat Generation Rate

Equation 1 was developed by finding the amount of heat generated in the pin per gram of material. This was then multiplied by the mass of the pin in the MCNP model and was divided by the height of the pin in the model. This returned the linear heat rate. The pin averaged flux was determined by using a scaling factor developed from the benchmark case.

$$\bar{\phi}_{pin} = \frac{T_{4-Run}}{T_{4-Bench}} \bar{\phi}_{B-11} \qquad \text{Equation 2}$$

where: T_{4-Run} = Tally 4 from MCNP Run in Question
$T_{4-Bench}$ = Tally 4 from MCNP ATR Benchmark Case
$\bar{\phi}_{B-11}$ = B-11 Averaged Flux at Given Axial Location

The two flux tallies from MCNP can be combined into a factor F. This leads to a new q' equation.

$$\frac{\bar{\phi}_{B-11} F_{Scale} A_{XC} E_{gen} M_{pin}}{H_{pin}} = q' \qquad \text{Equation 3}$$

When this equation is solved parametrically, it is quickly realized that it needs several conversion factors to convert the result form a value in MeV/cm to kW/ft. A sample solution follows for run 1 (see Table 8) at an axial location of -1 inches.

$$\frac{\bar{\phi}_{B-11} F_{Scale} A_{XC} E_{gen} M_{pin}}{H_{pin}} = q'$$

$$\frac{(4.8393E+14 n/cm^2 s)(.235534)(1.23155 cm^2)(1.28441 MeV/g)(11.1076 g)}{1 cm} \ldots$$

$$\ldots \times (1.60217646\text{E-}13\,\text{J/Mev})(10^{-3} kW/W)(2.54 cm/in)(12 in/ft) = q'$$

$$9.767 kW/ft = q'$$

Table 9 contains the maximum linear heat rate at a centerline position of -1 cm, which is the axial location of the maximum flux.

Table 9: Linear Heat Rate for Each Run Performed

Run	Gap Diameter	Scaling Factor	He-3	Heat Generated (T6)	q'	q'
#	mm	-	Fraction	MeV/g	W/cm	kW/ft
1	0.5	0.235	0	1.284	320.60	9.77
2	0.5	0.221	0.5	0.998	234.41	7.14
3	0.5	0.210	1	0.789	175.63	5.35
4	1.0	0.243	0	1.406	362.76	11.06
5	1.5	0.253	0	1.558	417.44	12.72
6	0.1651	0.257	0	1.634	445.27	13.57
7	2.0	0.277	0	1.491	437.53	13.34
8	0.5	0.224	0	1.642	390.12	11.89
9	1.0	0.201	1	0.583	124.02	3.78
10	1.5	0.195	1	0.481	99.51	3.03
11	0.1651	0.193	1	0.448	91.73	2.80
Neon	0.5	0.235	0	1.285	320.64	9.77
Half Density	0.5	0.235	0	1.284	320.62	9.77

DISCUSSION

The CASMO-3 output data is in line with expectations for a BWR pin cell depleted to those conditions. It includes both heavy metals formed from neutron capture and decay and fission isotopes present in an operating BWR pin. The inclusion of all of the isotopes in the BWR pin cell burn-up is important because, even if the isotopes were not present as soon as irradiation started (due to previous decay), they would soon form as the fission reaction proceeded in the BWTL. Since the pin was depleted to 25 GWd/MTU, it showed a large drop in U-235 in the pin cell. This drop in the amount of U-235 makes it very difficult for the fuel pin to generate the desired amount of heat. There are also large numbers of poisons present in the pin, making it even more difficult.

After benchmarking the neutron flux tallies against the linear heat generation rates summarized in Table 9, it can be seen that the values returned are within the range of current operating BWR reactors and well below design limits, which are approximately 25 kW/ft. The maximum value found was in run 6, which generated 13.57 kW/ft. This is within 10% from the desired value of 15 kW/ft. Run 8 demonstrates that by slightly altering the outermost dimensions of the test loop, a gas gap of only .5 mm would lead to a linear heat rate of 11.89 kW/ft. This is much higher than run 1, which has a gas gap of the same width. This makes it clear that altering the dimensions of the loop slightly could greatly affect the amount of heat generated in the fuel pin. Clearly, more runs, with many different dimensions, need to be performed to find the best mixture between neutronics and mechanical considerations. There seems to be no reason why a configuration with a linear heat rate of 15 kW/ft could not be found given more time.

An interesting trend in the data is found in how the altering the size of the gas gap on the exterior of the loop alters both the heat rate and the flux. As the gap size increases, the heat generation rate also increases. This trend can be seen in runs 1, 4, 5 and 6, all of which have the same composition. This trend is probably resultant from a change in the neutron scattering behavior of the loop, decreasing the amount of stainless steel present and placing something with more void in its place.

If run 1 is compared to the runs with the neon gas fill and helium gas fill at half density, it can be seen that the neutronic characteristics of the loop are not greatly altered by change in composition. This is because none of these gases are neutron absorbers, unlike He-3.

It can also clearly be seen in all of the runs performed at the same gas gap width that the linear heat rate drops significantly as He-3 is added to the system. In comparison to the maximum heat rate produced in run 6, the same loop conditions, given a He-3 concentration of 100% in the gas gap, would lead the heat generation rate in the entire loop to drop significantly to 2.80 kW/ft. This is well below the desired 6 kW/ft steady state operating conditions. However, this not an issue since the He-3 could be diluted in order to reach the desire heat rate.

Uncertainties in the numbers found lie in some of the cards that were used in the MCNP input deck. This includes the NONU card and the definition of the source. More research needs to be done to verify whether or not the NONU card needs be included in the input deck. The definition of the source as a single plane also needs to be checked and verified by others. By far the best way to calculate the values needed would be to insert the materials and surface cards into a model of the ATR core and then find the values desired. Given more time, a more adequate benchmark calculation could be performed and verified to a greater extent.

CONCLUSION

This project involved coupling a CASMO-3 fuel pin burn-up calculation with an irradiation simulation in MCNP5. The goals of the project were to develop the burn-up, determine the linear heat rate of the fuel pin in the ATR reactor and then simulate power ramping with He-3. The objectives summarized for this project were met but need to be verified further. The first goal to develop a CASMO generated burn-up was met and verified against expected values. The maximum linear heat rate for the BWTL pin was slightly lower than the desired value. It was 13.57 kW/ft and not 15 kW/ft. To compensate for this, variations could be made by slightly altering the geometry of the test loop, which would influence neutron scattering in the regions of concern. If 13.57 kW/ft is taken as the peak linear heat rate, then given the linear relation between total core power and flux in the B-11 position the power of the reactor would have to be raised by roughly 10% to 120 MWt to reach the desired 15 kW/ft. However, since conditions in the ATR can vary greatly, flux calculations at various higher lobe powers would need to be performed to give an accurate value to raise linear heat rate to the desired value. In all runs performed, diluting the gas gap with He-3 resulted in the power in the pin dropping to the desired goal of 6 kW/ft or lower, which proves that it is possible to perform ramp tests. The 13.57 kW/ft maximum value occurred with a gas gap of 1.651 mm, which is higher than the desired value of .5 mm. At this value the maximum linear heat rate is only 11.89 kW/ft.

REFERENCES and WORKS CITED

[1] Edenius, M. and Forssen, B., 1989, CASMO-3 A FUEL ASSEMBLY BURNUP PROGRAM User's Manual, Studsvik AB.

[2] X-5 Monte Carlo Team, 2008, *MCNP — A General Monte Carlo N-Particle Transport Code, Version 5*, Los Alamos National Laboratory

[3] Kim, S. and Schnitzler, B., 2008, "ADVANCED TEST REACTOR: SERPENTINE ARRANGEMENT OF HIGHLY ENRICHED WATER-MODERATED URANIUM-ALUMINIDE FUEL PLATES REFLECTED BY BERYLLIUM" *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, Nuclear Energy Agency.

[4] Hochreiter, L. E., Egrun, S. and Robinson, G. E., 2008, *Elements of Nuclear Reactor Design*, Class Notes for Nuclear Engineering 430 at The Pennsylvania State Univ., University Park, PA.

[5] Idaho National Laboratory: Advanced Test Reactor National Scientific User Facility, 2009, *Advanced Test Reactor National Scientific User Facility User's Guide*.

APPENDIX A

The CASMO-3 input deck for the BWR pin cell depletion calculations is summarized in this appendix.

```
TIT * 25 GWD/MTU BURNUP OF BWR PIN CELL
TFU=1060 TMO=555 VOI=42.6 BOR=0 * HOCHREITER 2-107, 5-45
FUE 1 10.32/4.401 * HOCHREITER 2-107
PIC .52832 .53974 .62611 .91715/'1' 'AIR' 'CAN' 'COO' * HOCHREITER 2-107
PDE 25.9 * HOCHREITER 2-107
PRE 71.71 * HOCHREITER 2-107
DEP -25.0 * USER SPECIFIED
STA
END
```

APPENDIX B

A sample MCNP5 input deck for the BWTL test loop is included in this section. Please not that the sections marked '***ALT***' were altered in the various runs performed to provide multiple data points. It is also important to mention that much of the formatting of the code was altered in transporting it into Microsoft Word. This section also includes the benchmark MCNP run, which was used to develop a scaling factor for the flux.

```
Pin Cell in ATR B11
c
c  By: Nathan Andrews
c      Department of Nuclear Engineering
c      The Pennsylvania State University
c ==============================================================================
c Cell lattice
1   1   -10.32   -1 5 -6       imp:n,p=1        $ Fuel Pin
2   2   -0.001         1 -2 5 -6     imp:n,p=1        $ Gap He/Air
3   3   -6.506   2 -3 5 -6     imp:n,p=1        $ Cladding Zircaloy-2
4   4   -0.445   3 -4 5 -6     imp:n,p=1        $ Coolant Inner Boiling
5   5   -8.027   4 -11 5 -6    imp:n,p=1        $ SS Inner Flow Tube
6   6   -.769    11 -12 5 -6   imp:n,p=1        $ Coolant Outer No Boil
7   5   -8.027   12 -13 5 -6   imp:n,p=1        $ SS Outer Flow Tube
8   7   -.01     13 -14 5 -6   imp:n,p=1        $ Helium Gap (cool T, loop P)
9   5   -8.027   14 -15 5 -6   imp:n,p=1        $ SS Pressure Tube
10  8   -.98433  15 -16 5 -6   imp:n,p=1        $ ATR Primary Coolant
11  0           16 5 -6 7 -8 9 -10   imp:n,p=1   $ Inner Void and Source Line
12  0           -5:6:-7:8:-9:10      imp:n,p=0   $ Graveyard

c ==============================================================================
c Surface cards
1   cz    0.52832   $ Pellet external radius
2   cz    0.53974   $ Clad internal radius
3   cz    0.62611   $ Clad external radius
4   cz    0.95885   $ Inner Coolant Outer Radius UPWARD (Boiling)
11  cz    1.0033    $ Inner Flow Tube Outer Radius
12  cz    1.23125   $ Outer Coolant Outer Radius DOWNWARD (Non-Boiling)
13  cz    1.3335    $ Outer Flow Tube Ouer Radius
14  cz    1.4335    $ Helium Gap Outer Radius           ***ALT****
15  cz    1.7399    $ Pressure Tube Outer Diameter       ***ALT*** .065 in gap
16  cz    1.905     $ ATR Outer B-11 Position
*5   pz   -0.5      $ Bottom
*6   pz    0.5      $ Top
*7   px    -2   $ Left X
*8   px     2   $ Right X
*9   py    -2   $ Left Y
*10  py     2   $ Right Y
```

```
c ================================================================================
c Data cards:
SDEF   PAR=1  ERG=d1 X=-2 Y=d2 Z=d3 VEC 1 0 0 DIR 1  $Source
SI1 H 1.0E-10 1.0E-09 1.0E-08 1.0E-07 1.0E-06 1.0E-05 1.0E-04 1.0E-03
      1.0E-02 1.0E-01 1. 14.   $Energy Regions
SP1 D 0 .00016 .01725 .31204 .10845 .06147 .06882 .07441 .07582 .08461
      .12166 .07532  $ Relative Probability
SI2 -1.905 1.905
SP2 0 1
SI3 -.5 .5
SP3 0 1
MODE N P            $ Neutron and Photon Mode
nps 100000
PRINT 110
NONU
c ================================================================================
c Material composition
c Fuel pin = UO2 (T=900K - Operating Temp)
m1    8016.72c     6.70756E-01           $ T=900
      92234.72c     7.89622E-05           $ T=900
      92235.72c     7.80770E-03           $ T=900
      92236.72c     1.44782E-03           $ T=900
      92238.72c     3.13335E-01           $ T=900
      92239.72c     1.61648E-07           $ T=900
      93237.72c     1.44523E-04           $ T=900
      93239.72c     2.33249E-05           $ T=900
      94238.72c     3.78136E-05           $ T=900
      94239.72c     2.91955E-03           $ T=900
      94240.72c     5.35788E-04           $ T=900
      94241.72c     3.98317E-04           $ T=900
      94242.72c     5.16177E-05           $ T=900
      95241.72c     1.30344E-05           $ T=900
      95242.72c     2.30913E-07           $ T=900
      95243.72c     8.93868E-06           $ T=900
      96242.72c     2.60982E-06           $ T=900
      96244.72c     1.83424E-06           $ T=900
      36083.72c 3.40737E-05           $ T=900
      45103.72c 2.45877E-04           $ T=900
      45105.72c 5.56159E-07           $ T=900
      47109.72c 2.57571E-05           $ T=900
      54131.72c 2.02416E-04           $ T=900
      55133.72c 5.09362E-04           $ T=900
      55134.72c 4.83912E-05           $ T=900
      54135.72c 1.89973E-07           $ T=900
      55135.72c 3.15979E-04           $ T=900
      60143.72c 4.19264E-04           $ T=900
      60145.72c 2.84935E-04           $ T=900
      61147.72c 8.15721E-05           $ T=900
      62147.72c 3.16572E-05           $ T=900
      61148.72c 3.81413E-07           $ T=900
      62148.72c 9.27064E-07           $ T=900
      62149.72c 3.27665E-06           $ T=900
      62150.72c 1.15836E-04           $ T=900
      62151.72c 1.24441E-05           $ T=900
      62152.72c 4.08381E-05           $ T=900
```

```
     63153.72c 4.21550E-05          $ T=900
     63154.72c 1.24798E-05          $ T=900
     63155.72c 6.99972E-06          $ T=900
     53135.72c 2.30538E-07          $ T=900
     61149.72c 5.08300E-07          $ T=900
     64155.72c 1.51961E-07          $ T=900
c Gap = Oxygen (From CASMO Default)
m2    8016.72c   1              $ Oxygen (900K)
c Cladding = Zircaloy-2
m3    40090.71c -0.9825         $ Zr-40 (600K)
      50112.71c -0.0145         $ Tin (600K)
      24052.71c -0.0010         $ Cr (600K)
      26056.71c -0.00135        $ Iron (600K)
      28062.71c -0.00055        $ Nickel-62 (600K)
c Moderator = H20 (Inner)
m4    1001.53c   0.6667     $ H (587.2K)
      8016.53c   0.3333     $ O16 (587.2K)
mt4   lwtr.62t              $ Thermal scattering (600K)
c SS 316 L (ASTM A240)
m5    6012.42c    -.00015    $ C (300K)
      25055.71c   -.0100     $ Mn (600k)
      14028.71c   -.00375    $ Si (600k)
      24052.71c   -.17       $ Cr (600k)
      28059.71c   -.1200     $ Ni (600k)
      42096.71c   -.025      $ Mo (600k)
      15031.71c   -.000225   $ P (600k)
      16032.71c   -.00015    $ S (600k)
      7014.71c    -.0005     $ N (600k)
      26056.71c   -.670225   $ Fe (600k)
c Moderator = H20 (Outer)
m6     1001.53c   0.6667     $ H (587.2K)
       8016.53c   0.3333     $ O16 (587.2K)
mt6    lwtr.62t              $ Thermal scattering (600K)
c Helium (Gap)   ***ALT***
m7     2003.66c  1   $ He-3 (293K)
c      2004.66c  0.6667   $ He-4 (293K)
c Coolant = H20 (ATR B11)
m8     1001.42c   0.6667     $ H (300K)
       8016.42c   0.3333     $ O16 (300K)
mt8    lwtr.60t              $ Thermal scattering (294K)
c ==========================================================================
c Tally cards
F6:N,P 1 2 3 T
```

```
Base Benchmark in ATR B11
c
c  By: Nathan Andrews
c     Department of Nuclear Engineering
c     The Pennsylvania State University
c ==========================================================================
c Cell lattice
1  1  0.099591     -1 5 -6              imp:n,p=1 $ B11 H2O annulus
2  2  1.2387-1      1 -2 5 -6           imp:n,p=1 $ B11 Be filler
3  3  0.06000038    2 -3 5 -6           imp:n,p=1 $ B11 Al clad
4  1  0.099591      3 -4 5 -6           imp:n,p=1 $ B11 H2O annulus
```

```
11  0          4 5 -6 7 -8 9 -10    imp:n,p=1   $ Inner Void and Source Plane
12  0          -5:6:-7:8:-9:10      imp:n,p=0   $ Graveyard

c ===========================================================================
c Surface cards
1 cz 0.31750 $ B11 water annulus
2 cz 1.62687 $ B11 Be filler
3 cz 1.82499 $ B11 Al clad
4 cz 1.90500 $ B11 water annulus
*5   pz   -0.5      $ Bottom
*6   pz    0.5      $ Top
*7   px    -2   $ Left X
*8   px    2    $ Right X
*9   py    -2   $ Left Y
*10  py    2    $ Right Y

c ===========================================================================
c Data cards:
SDEF   PAR=1  ERG=d1 X=-2 Y=d2 Z=d3 VEC 1 0 0 DIR 1  $Source
SI1 H 1.0E-10 1.0E-09 1.0E-08 1.0E-07 1.0E-06 1.0E-05 1.0E-04 1.0E-03
      1.0E-02 1.0E-01 1. 14.   $Energy Regions
SP1 D 0 .00016 .01725 .31204 .10845 .06147 .06882 .07441 .07582 .08461
      .12166 .07532  $ Relative Probability
SI2 -1.905 1.905
SP2 0 1
SI3 -.5 .5
SP3 0 1
MODE N P            $ Neutron and Photon Mode
NPS 2000000
PRINT 110
NONU
c ===========================================================================
c Material composition
c Beryllium 1.8538 g/cm3
c 100% Be
m1 4009.50c 1.2387-1
mt1 be.01t
c water
c 1005, 0.9931 g/cm3.
m2   1001.50c 6.6394-2
    8016.50c 3.3197-2
mt2 lwtr.01t
c
c Al Side Plate 2.7 g/cm3
m3   14000.50c 3.71E-04
     26000.55c 1.16E-04
     29000.50c 6.40E-05
     25055.50c 2.37E-05
     12000.50c 6.02E-04
     24000.50c 5.94E-05
     29000.50c 7.46E-06
     22000.50c 6.79E-06
     13027.50c 5.88E-02
c ===========================================================================
c Tally cards
F4:N 1 2 3 4 T
```

CURRICULUM VITAE

NATHAN C. ANDREWS
nca5010@psu.edu

150 Gregg St
Spring Mills, PA 16875
Home: (814) 422-0345
Cell: (814) 574-6681

Education:	*Massachusetts Institute of Technology, Cambridge, MA* Graduate Work in Nuclear Engineering Will Begin Research June 2010 Will Begin Classes September 2010 *The Pennsylvania State University, University Park, PA* B.S. in Nuclear Engineering *St. Petersburg State University, Russia* Intensive Russian Language Instruction at Smolny Institute Summer 2008 *Moscow Humanities Institute, Russia* Intensive Russian Language Instruction Summer 2008
Academic Honors:	*Schreyer Honors College* Have been a member of the prestigious honors college at The Pennsylvania State University since first semester. *American Nuclear Society Undergraduate Scholarship* Received both freshman and undergraduate scholarship awards. *Deans List* Every semester. *Tau Beta Pi Engineering Honors Society* Inducted spring 2009. *Exelon Fellowship* Received departmental scholarship for academic performance in nuclear engineering. *Louis Harding Memorial Scholarship* Received departmental scholarship for academic performance while pursuing concurrent majors in mechanical and nuclear engineering.
Research:	*Nuclear Engineering Honors Thesis* • Working in co-ordination with the Advanced Test Reactor National Scientific Users Facility at the Idaho National Lab • Determined the linear heat generation rate of a burned BWR fuel segment in the ATR B-11 position.

	• Developed the burn-up of a BWR fuel pin to 25 GWD/MTU in CASMO-3 and coupled the output to MCNP5 to determine the linear heat generation rate.
Technical Skills:	• Experience in programming in Fortran and Matlab • Proficient in computer aided design using Solidworks • Class work with COBRA, ANC, CASMO-3 and MCNP5.
Work Experience:	*Nuclear Operations Intern, Idaho National Laboratory* • Designed basic in core system of a boiling water test loop for the Advanced Test Reactor • Summer 2009 • Learned the fundamental behinds the operation of the Advanced Test Reactor • Used both reactor simulator and the loop test simulator • Received Radiological Worker II training and made entry into contamination areas. • Performed system walk downs of both secondary and primary systems.
Activities/ Community Involvement:	*The Pennsylvania State University – Varsity Wrestling Team* Member of the wrestling team from 2006 to 2009. • Varsity Letterman, 2008-2009 • Big Ten Distinguished Scholar Award, 2008-2009 *Student Senator Engineering Department* Elected to the Student Assembly and Faculty Senate *Engineering Undergraduate Council* *American Mensa Society* *American Society of Mechanical Engineers* *American Nuclear Society*