The Design of a High Temperature Gas Reactor Fuel Testing Facility For MITR-II

by

Christine Marie Martin

Submitted to the Department of Nuclear Engineering
In Partial Fulfillment of the Requirements
For the Degrees of

Bachelor of Science

and

Master of Science in Nuclear Engineering

at the

Massachusetts Institute of Technology

June 1991

© Massachusetts Institute of Technology 1991

Signature of Author

Department of Nuclear Engineering
May 14, 1991

Certified by

Lawrence M. Lidsky
Professor of Nuclear Engineering
Metcalfe Professor of Engineering and the Liberal Arts

Certified by

David D. Lanning
Professor of Nuclear Engineering

Accepted by

Allan F. Henry
Chairman, Department Committee on Graduate Students

JUL 12 1991
ARCHIVES
The Design of a High Temperature Gas Reactor
Fuel Testing Facility For MITR-II

by

Christine Marie Martin

Submitted to the Department of Nuclear Engineering
on May 14, 1991 in partial fulfillment of the
requirements for the Degrees of Bachelor of Science
and Master of Science in
Nuclear Engineering

Abstract

A facility for irradiating coated particle fuel under typical gas reactor conditions was
designed for the MIT research reactor. The test facility uses a combination of an electrically
heated graphite cylinder and nuclear heating to maintain the fuel temperature in the
operating range for gas reactors, up to 1000°C. The facility operates in a stagnant helium
atmosphere. Samples containing up to 50 coated particles may be accommodated.

The facility, designed for insertion into the 6 inch radial, horizontal port, will place the fuel
in the heavy water reflector near the reactor core tank. This location provides a relatively
high neutron flux without the safety-related restrictions which would be present if the
facility were to be inserted directly into the reactor core.

A combination of analytical solutions and FEM modelling was used to develop a system
capable of maintaining 1000°C fuel temperatures while not boiling the heavy water 38 mm
away. Absorption of neutrons and gamma rays within the facility was found to be an
important heat source. Neutron and gamma heating estimates were made based on
experimental data and Monte-Carlo simulation, and the facility was designed to remain
below 300°C when electric power is removed. With the reactor at 5 MW, 250 watts of
electrical power will be required. Approximately 300 watts will be required when the
reactor is shut down.

Thesis supervisors:

Lawrence M. Lidsky  Title: Professor of Nuclear Engineering
                    Metcalfe Professor of Engineering and the Liberal Arts
David D. Lanning  Title: Professor of Nuclear Engineering
Acknowledgements

While an undergraduate at MIT, one of the most important lessons I learned was that working with people helps you reach beyond your own limits. This lesson served me well as I worked on this thesis. Many people helped me along the way, and I want to use this opportunity to thank them.

Professors Lanning and Lidsky guided me through this work. When my design was weak, they pointed out the weakness and suggested how to strengthen it. Professor Lidsky has taught me to recognize good engineering, and he has taught me what it means to be a good engineer, beyond the technical skills. His integrity and strong sense of value have earned him my eternal respect. I will always aspire to live to the standards he has shown me.

Everett Redmond provided the model of the MIT reactor which supplied me with invaluable heating and neutronics information. He was always available to answer questions and explain his very complex model to me. I am deeply indebted to him. His support also is greatly appreciated.

The one person who has helped me the most is my husband Jerry. His help included ideas when I was stuck and encouragement when I was making progress. My time as a graduate student was very happy because I was able to spend it with him.

This research was performed under appointment to the Nuclear Engineering & Health Physics Fellowship Program administered by Oak Ridge Associated Universities for the U.S. Department of Energy.
# Table of Contents

Abstract ........................................................................................................................................... 2
Acknowledgements ...................................................................................................................... 3
Table of Contents ........................................................................................................................ 4
List of Figures .............................................................................................................................. 6
List of Tables ................................................................................................................................. 7

1. Introduction ................................................................................................................................ 8

2. HTGR Fuel ................................................................................................................................ 10
   2.1. Particle Fuels ................................................................................................................... 10
   2.2. The switch from BISO to TRISO .................................................................................... 12
   2.3. Particle Manufacture ....................................................................................................... 13
   2.4. Historical Failure Mechanisms ...................................................................................... 13
       2.4.1. Pressure Vessel Failure ......................................................................................... 13
       2.4.2. The Amoeba Effect ................................................................................................. 14
       2.4.3. Fission Product Attack of the SiC Layer ................................................................. 15
   2.5. Current Fuel Modelling .................................................................................................... 18
       2.5.1. Nabielek Silver Release Model .............................................................................. 19
       2.5.2. Grubmeier SiC Failure Model ................................................................................ 20
       2.5.3. Short-Lived Fission Gas Release Model ............................................................... 21
   2.6. Predictive Failure Models ............................................................................................... 21
       2.6.1. Statistical Interpretation of Experimental Results ............................................... 22
       2.6.2. Kaae Pressure Vessel Failure Model .................................................................... 22
       2.6.3. Goodin-Nabielek SiC Model .................................................................................. 23
       2.6.4. Martin-Goodin-Nabielek Failure Model ............................................................... 24
   2.7. Future Directions for Fuel Development ........................................................................ 25

3. System Overview ....................................................................................................................... 27
   3.1. Design Goals ................................................................................................................... 27
   3.2. Design Constraints .......................................................................................................... 28
   3.3. The TRIGA King Furnace Facility .................................................................................. 30
   3.4. Interfacing with MITR-II .............................................................................................. 31
   3.5. Sample Design and Manufacture ................................................................................... 35
   3.6. Heat Source .................................................................................................................... 36
   3.7. Control and measurement systems ............................................................................... 37
   3.8. Sample Handling ........................................................................................................... 38

4. Neutronics .................................................................................................................................. 39
   4.1. Description of MCNP .................................................................................................... 40
   4.2. The MCNP Model of the MITR-II .............................................................................. 40
   4.3. MCNP Results and Analysis ......................................................................................... 48

5. Thermal Design ....................................................................................................................... 55
   5.1. Heater configuration ....................................................................................................... 56
   5.2. Transitions from high temperature to low temperature .................................................. 59

4
5.3. Analysis tools ............................................................................. 63  
5.3.1. Analytical approximations ..................................................... 63  
5.3.2. ALGOR® Computer Code ......................................................... 68  

6. Material choices............................................................................ 75  
6.1. Al-6061 ..................................................................................... 76  
6.2. Graphite .................................................................................... 77  
6.3. Insulation ................................................................................... 78  

7. Control and Measurement.............................................................. 79  
7.1. Temperature ............................................................................. 79  
7.2. Atmosphere ............................................................................... 80  

8. Sample Handling............................................................................ 82  
8.1. Activity of sample after irradiation ........................................... 82  
8.2. Inserting and removing samples ............................................... 85  

9. Future Work.................................................................................. 88  
9.2. Further work on space external to the port .............................. 90  
9.3. Related Facilities ...................................................................... 90  

10. Conclusions.................................................................................. 92  

A. Burn-up Estimates....................................................................... 93  

B. Estimation of Heat Transfer Coefficients ..................................... 95  
B.1. Heat Transfer Coefficient for Forced Convection .................... 95  
B.2. Natural Convection Heat Transfer Coefficient .......................... 96  

C. Heat Loss through the Hemisphere at the end of the Port ........... 99  

References....................................................................................... 102
List of Figures

Figure 2.1 TRISO coated fuel particle 12
Figure 2.2 Palladium and cesium concentrations as a function of radial position in a test particle 17
Figure 3.1 TRIGA King Furnace 32
Figure 3.2 6RH2 port cross-section 34
Figure 4.1 Axial view of MCTR-II as modelled in MCNP 43
Figure 4.2 Heating estimates by Boerigger and Zaker 44
Figure 4.3 Model of Boerigger's experiment for MCNP 45
Figure 4.4 Cross-sectional view of the MCNP model through the core and the 6RH2 port 46
Figure 4.5 Cross-section of the fuel test facility as modelled in MCNP 47
Figure 4.6 Heating rates as calculated by Boerigger and as predicted by MCNP 50
Figure 4.7 Axial heating profile in the Fuel Test Facility 51
Figure 4.8 Radial profile of heating by neutrons and gamma rays in the heater section 52
Figure 5.1 Electric circuit of heater, schematic 58
Figure 5.2 Tip of facility, with "fin" section 61
Figure 5.3 Tip of facility with graphite extension piece 62
Figure 5.4 Heat dissipated by the fin, and the temperature 0.2 m from the fin's base 66
Figure 5.5 Temperature profile, electric heating of 300 watts, no neutron and gamma heating 71
Figure 5.6 Temperature profile, electric heating of 250 watts, neutron and gamma heating from reactor at 5MWth 72
Figure 5.7 Temperature profile, no electric heating, neutron and gamma heating from reactor at 5MWth 73
Figure 5.8 Magnitude of the heat flux, for the case of 250 watts electric power, neutron and gamma heating from MCTR-II at 5MWth 74
Figure 8.1 Activity of one particle as a function of cooling time and burn-up 85
Figure 8.2 Sample and sample removal tool 87
Figure C.1 Schematic of hemisphere for heat transfer 99
List of Tables
Table 3.1 Typical neutron fluxes in AVR 28
Table 4.1 Neutron flux at the center of the heater, compared to flux levels in AVR 53
Table 5.1 Electrical resistance per unit length in different parts of the electrical heating circuit. 57
Table 7.1 Impurity levels in reactor helium used by Bates et al. 81
Chapter 1

1. Introduction

Advanced nuclear fuels may be the key to incorporating passive safety in the next generation of nuclear reactors. An example is the Modular Gas-cooled Reactor (MGR), which relies on the ability of coated particle fuel to withstand very high temperatures. Coated particle fuel can withstand temperatures up to 1600°C for hundreds of hours without releasing biologically significant fission products.[N2] These high fuel failure temperatures allow the design of reactors in which the fuel does not reach its failure temperature during an accident, even without active safety systems and cooling. Since the safety of MGR's relies in large part on fuel performance, it is important that we be able to predict the fuel behavior under accident conditions. Fission product release from particles has been studied for many years with the goal of predicting fuel performance based on first principles. The failure mechanisms are not yet fully understood, so fuel behavior predictions now are based on statistical failure data collected from experiments. It is the goal of this thesis to present the design of an experimental facility to irradiate coated particle fuel. This facility will support experiments which will lead to an understanding of the failure mechanisms of coated particle fuel.

The experimental facility, the Fuel Test Facility, is designed to be inserted in a port of the MIT reactor. The facility will make it possible to irradiate small quantities of coated particle fuel at temperatures up to 1000°C in a helium atmosphere.

The reactor at MIT, the MITR-II, can provide the neutron flux necessary to irradiate coated particle fuel, creating the fission products whose transport we wish to study. Using the MIT reactor to perform these irradiations will give experimenters the opportunity to
directly control the experiments. If facilities are constructed to produce the experimental fuel, irradiate it, and examine it, the whole process can be much faster than if some of the work needs to be done off-campus. The MIT reactor offers the potential for a small experiment which will be more responsive to changing lines of investigation, unlike large scale tests which have been run to gather statistical data.

There are a number of interesting issues which the proposed facility will be able to address. Fuel failure mechanisms can be investigated, with the goal of developing phenomenological models of the fuel to be used for performance prediction. In the process of learning what the failure mechanisms for coated particle fuel are, knowledge will be gained which may be used to design improved coatings for the fuel.
Chapter 2

2. HTGR Fuel

2.1. Particle Fuels

The Modular High Temperature Gas-cooled Reactor (MHTGR) relies on the very high temperature capability of its coated particle fuel to retain the fission products even during a loss of coolant accident. The coated particle fuel used in MHTGR's can retain fission products at up to 1600°C. Extremely low fission product release rates during normal operation result in the additional benefit of low occupational exposures.

Coated particle fuels are based on microspheres of fissile or fertile material, with ceramic coatings. The outer dimension of each particle is approximately 900 μm. Hundreds of particles are held together by a carbon-based binder which is formed into rods (about 13 mm diameter), or spheres (about 60 mm diameter). In the core of a large nuclear reactor, there can be as many as $10^{11}$ coated fuel particles. Coated particle fuels were first used in the Peach Bottom reactor in the United States, AVR in the Federal Republic of Germany, and in the Dragon Project in England in the 1960's.

These particles are designed to act as individual pressure vessels to retain fission products. Reactors using this fuel rely on the particle coatings as their primary fission product boundary. For this reason, the ability of particles to retain fission products is of great importance in the licensing of MHTGR's. The causes of fission product release from particles are being studied, with the goal of predicting the fission product release during normal and transient operating conditions. In 1977, Gulden and Nickel had this to say about the work in this area, "... rapid progress is being made, and it appears likely that
performance calculations from first principles will be feasible in the future.” In 1991, their prediction has not yet been fulfilled.

**Coated Particle Fuel Description**

As mentioned above, coated particle fuel is based on microspheres of fissile or fertile material. The most common materials used are uranium oxides and uranium carbides, although particles made with thorium and plutonium are possible. The particle consists of a sphere of fissile or fertile material approximately 500 μm in diameter. This sphere is called the “kernel”. As the name suggests, the kernel is coated— with layers of ceramics intended to provide retention of fission products at the particle level. The first layer surrounding the kernel is the “buffer” layer. The buffer layer is a low-density pyrolitic carbon layer 90 μm thick which serves to slow the recoil of fission products, and provides room for the expansion of fission gases. Two types of coatings have been used on top of the buffer. The first, termed BISO, had just a layer of high density pyrolitic carbon intended to act as a miniature pressure vessel to retain the fission products. TRISO fuel was developed later. TRISO fuel has three layers on top of the buffer layer— an inner high density pyrolitic carbon layer 40 μm thick, a silicon carbide layer 40 μm thick, and an outer high density pyrolitic carbon layer 40 μm thick. In TRISO fuel, the silicon carbide layer provides the strength for pressure vessel, as well as providing a layer with very low diffusivity. The inner pyrolitic carbon serves mainly as a shield against fission products for the silicon carbide. The outer pyrolitic carbon layer supports the silicon carbide layer and protects it from scratches which would act as stress concentrators. Figure 2.1 shows a diagram of the cross-section of a TRISO particle.
2.2. The switch from BISO to TRISO

Silicon carbide has many features which make it suitable for application in coated particle fuel. It has a low absorption cross section for neutrons and high thermal conductivity, two important features for fuel cladding. SiC has a small interatomic spacing, resulting in low diffusion rates through it by fission products. For example, in anisotropic SiC deposited at or above 1500°C, the diffusion constant for cesium at 1400°C is less than $6 \times 10^{-21} \text{ m}^2/\text{s}$ and for strontium, also at 1400°C, it is less than $5 \times 10^{-17} \text{ m}^2/\text{s}$.[P1] An added benefit of the SiC layer is that it reduces the migration of uranium during the manufacturing process, so there is less contamination of the outer pyrocarbon.[N2]

The main reason to include an inner pyrocarbon layer before the SiC layer is to protect the kernel from chlorine present during the SiC deposition. In addition, silicon carbide is
susceptible to attack by fission products, so it must be separated from the buffer layer (in which fission products are unrestrained) by a layer of dense pyrocarbon. To protect the outside of the silicon carbide and provide mechanical support, an additional dense pyrocarbon layer is added. This four layer structure is anomalously termed a TRISO particle. BISO particles are no longer used for fissile kernels, although they may be used for fertile kernels to aid reprocessing.

2.3. Particle Manufacture

The coatings are deposited on the kernels in a spouted bed coating furnace. The kernels are suspended in a gas stream by the force of the gas flow. At the temperature of the furnace, the gas used decomposes and forms agglomerates, which build up on the surface of the particle. There are a number of independent variables in this process, including the type of gas, the deposition temperature and the deposition rate of the layer. For the deposition of the pyrocarbon layers, methane gas and propane gas are commonly used, while silane gas (CH$_3$SiCl$_3$) is used to deposit the SiC layer.

Particles can be categorized by the temperature of the pyrolytic carbon coating process as well as the type of coating. The three types of pyrolytic carbons are BISO low temperature isotropic (LTI), BISO high temperature isotropic (HTI), and TRISO LTI. The high temperature coating retains solid fission products (especially cesium) better than the low temperature coating. For BISO particles this difference is significant, but for TRISO particles, retention by the SiC layer is the most important, so the low temperature process can be used to deposit the pyrocarbon, and still retain an excellent diffusion barrier for solid particles.[H4]

2.4. Historical Failure Mechanisms

2.4.1. Pressure Vessel Failure

The failure mechanism which was originally thought to be life-limiting is a stress-induced failure of the coating layers, termed pressure vessel failure. Fission gases
produced in the kernel during irradiation increase the internal pressure of the particle, which leads to tensile stresses in the coating layers which may exceed the fracture stress. Among the factors influencing pressure vessel failure are: the irradiation temperature, which will determine the amount of the fission gases released from the kernel; the fast neutron fluence; burnup, which determines the amount of gaseous products produced; the dimensions of the kernel and layers; densities and the anisotropy of coating layers.[H4]

Particles typically operate in the temperature range from $800^\circ - 1200^\circ C$. This high temperature, in combination with the high fluence, may cause creep, which would help relieve some of the stresses in the coating. This creep is a factor which should be considered in failure models to accurately predict failure. Bennett, at the Idaho National Engineering Laboratory, is currently developing an elastic-plastic finite element method to handle this.[B2] One solution to pressure vessel failure involves improving the manufacturing techniques to minimize the range of the kernel diameter and the coating thickness. This will make pressure vessel failure more predictable, so fuel can simply be designed to withstand the buildup of pressure in expected burnup ranges.

Pressure vessel failure involves the fracture of a ceramic material, either pyrolitic carbon or silicon carbide. It is very difficult to define a unique fracture stress for a ceramic, due to the strong effect of defects in the material. This necessitates the use of Weibull statistics. Weibull statistics give a probability of failure for a given stress value. In addition to the uncertainty in the fracture stress, there is also variability in the coating thicknesses, the local density and the local degree of anisotropy, all of which affect the fracture stress. Nonetheless, in modern coated particle fuels, pressure vessel failure of undamaged particles makes a negligible contribution to total release.

2.4.2. The Amoeba Effect

Examination of early fuels showed that kernels migrated up temperature gradients, through the buffer layer, and thinned and weakened the pyrocarbon layer. This effect was
dubbed the amoeba effect, because in photomicrographs of particles, the kernel sometimes assumes an amoeba-like shape and appears to have moved like an amoeba from its initial position. The effect occurs in particles with a carbide kernel and particles with an oxide kernel, although the postulated mechanisms for these two types of kernels are quite different.

In a carbide kernel, a quantitative description can be based on solid-state carbon diffusion. Solid-state diffusion and gas phase transport are used to describe kernel migration of oxide fuels. The theory for oxide fuels is that carbon diffuses by solid-state diffusion in one direction, and carbon monoxide travels the opposite direction by gas phase transport.[H4] In oxide fuel, it has been assumed that the amount of free oxygen left after fissions plays an important role in this mechanism. Moving to hypostoichiometric fuel leaves less carbon monoxide available for participation in the mass transport of carbon, thereby suppressing the kernel migration in oxide fuels. Uranium oxycarbide kernels were developed by ORNL that are less susceptible to amoeba-type failure.[H4]

The thermal gradient in the fuel is an important factor in any theory about the amoeba effect. Wagner-Löffler compiled data from a number of experiments and compared the average temperature of the particles, the temperature gradient and the average displacement of the kernel.[W1] Thermal gradients in excess of 80 °C/cm seem to be necessary for the amoeba effect to take place.

2.4.3. Fission Product Attack of the SiC Layer

Rare-earth fission products migrate down temperature gradients, through the buffer and inner pyrocarbon layers, and concentrate at the inner pyrocarbon – silicon carbide interface. The rare-earth metals react with the silicon carbide, degrading the integrity of the silicon carbide layer.[M2] At high burnups, when the concentration of fission products is high, chemical attack by fission products on the silicon carbide layer is more important than diffusion. The chemical attack is concentrated both at flaws in the structure and at grain
boundaries, and is more rapid in silicon carbide layers deposited at lower temperatures.  

Investigators are currently looking at palladium in regard to this attack. Figure 2.2 shows the concentration of elements in a particle.

One possible factor in silicon carbide attack may be the presence of trace chlorine. This chlorine is a residue from the silicon carbide deposition. Chlorine forms chlorides with metallic fission products, which move more easily through the inner pyrocarbon layer, resulting in higher concentrations of these metals at the silicon carbide boundary. A solution to this problem requires an impervious inner pyrocarbon layer, to ensure that during the silicon carbide deposition, no chlorine can get into the kernel to aid movement of fission products later.

Once metallic fission products are at the SiC layer, attack can occur along the grain boundaries. Cesium and silver vapor will penetrate along the grain boundaries in pyrolitic silicon carbide. Another factor which influences the permeation rate by metallic fission products in pyrolitic silicon carbide is the amount of free silicon or porosity present. In silicon carbide deposited below 1400°C, the permeation by metallic fission products is much greater than in pore-free silicon carbide deposited at temperatures above 1500°C with no free silicon.
Figure 2.2. Palladium and cesium concentrations as a function of radial position in a test particle.[N3]
Some silicon carbide attack can be averted if the fission products which attack the SiC layer can be retained in the kernel. For strontium and barium in oxide kernels, there is little release from the kernel up to 1475°C.[H2] For temperatures between 1500°C and 1650°C, however, enough strontium and barium are released from the kernel to attack the silicon carbide. These metals can completely penetrate the silicon carbide. Once that has occurred, strontium, barium, cesium are no longer retained by the silicon layer.[H2] At very high temperatures, above 2000°C, silicon carbide thermally decomposes into silicon gas and graphite, and so its fission product retention ability vanishes.

The stoichiometry of the fuel kernel affects the issues of silicon carbide attack, kernel migration, and internal pressure. Thermodynamic calculations performed by Homan et. al., show the relationships between these failure mechanisms and the stoichiometry of UCO fuel. For fuel with less than 15% UC₂ (remainder UO₂), there is excessive carbon monoxide overpressure. For 100% UO₂, kernel migration and silicon carbide layer oxidation can be a problem. When there is greater than 70% UC₂, the rare earth metals do not oxidize, are not held in the kernel, and may attack the silicon carbide layer. These predictions were supported by irradiation of fuels.[H3]

2.5. Current Fuel Modelling

In the ideal case, enough tests would be run on fuel particles to determine their normal and accident performance to within acceptable levels of uncertainty. This requires a large number of particles to be tested over a wide range of conditions. In the absence of such an extensive database, models are needed to predict performance. Models are also needed to explain the phenomena discovered experimentally. Additionally, models can suggest directions for further fuel development.

One objective for model development is to use the model to determine a source term for probabilistic risk assessments. Another objective of models is to make a "realistic" prediction of fission product release and fuel degradation as a function of time and
temperature. [R1] Yet another objective for models is to quantify the effects of pre-accident conditions on fuel performance during an accident. Ideally, a model would be based only on physical theories for all relevant failure and transport mechanisms. Such a model would use physical parameters, not mathematical. [R1]

Currently, there is no one model which takes into account all variables which will affect fuel performance, but many models which can account for or predict behavior when only a few parameters are variable. The first category of models includes those “models” which offer qualitative descriptions of fuel failure. These models attempt to explain the mechanisms causing the failure.

2.5.1. Nabielek Silver Release Model

Experimental results show that after 10 days at 1500°C, silver was released from TRISO particles while cesium and strontium were not released after 210 days at this temperature. [N1] Silver oxide is unstable at high temperatures, so while cesium and other similar metals are held somewhat in the fuel kernel, silver is not held well. It is known that silver-110m is not retained by pyrocarbon, so the only barrier to silver release in a TRISO particle is the silicon carbide layer. The Nabielek model attempts to explain the transport of silver-110m through the silicon carbide layer.

Along with the information that there is almost total release of silver-110m after about 10 days at 1500°C, there is also information about a temperature dependance of release. There is a steep increase in the silver release between 1140° and 1240°C and “its occurrence in both the shorter and the longer duration experiment are conspicuous.” [N1] Nabielek’s proposed release mechanism involves the formation of a silver/silicon alloy with a melting point around 840°C which coexists with solid silicon. He postulates that the grain boundaries are permeable to this liquid alloy. When the silicon carbide has free silicon present, this alloy can form and provide a mechanism for the release of silver. Nabielek further predicts that even silicon carbide which has no measurable free silicon will have
enough free silicon to support this silver transport mechanism. So the Nabielek model predicts silver-110m release for all fuels we can currently manufacture.

2.5.2. *Grübmeier SiC Failure Model*

Grübmeier determined that chlorine presence and the type of kernel (carbide or oxide) affect the silicon carbide layer’s stability.[G3] Under irradiation, SiC corrosion occurs on the cold side of the particle. Therefore, an explanation based on fission product transport by carbide migration is incorrect, since migration should lead to concentration of the fission products on the hot side of the particle and corresponding SiC attack on the hot side, not the cold side as experimental results show. This rules out a dependance on solid-state diffusion of fission products.

For carbide kernels, there is a weak bond formed between lanthanides and carbon. This allows the lanthanides to diffuse through the inner pyrocarbon layer. When the lanthanides reach the silicon carbide layer, the lanthanides react with the silicon carbide to form silicides. For oxide kernels, when there is no chlorine present, there is no silicon carbide attack. When chlorine is present, the silicon carbide is attacked. The determination that chlorine is the driving factor in oxide kernels is drawn from a large number of irradiations performed at research reactors in Jülich and other locations. Analysis of particles from these irradiations showed that in the area of silicon carbide attack, rare earth metals and chlorine were present. Grübmeier theorized that the chlorine bonded with the rare earth metals, and they moved together as metal chlorides more freely than the rare earths would have moved alone.

There is not enough information on the amount of chlorine present in the particles at the onset of corrosion of the silicon carbide layer to set an absolute limit on the allowable amount of chlorine in particles.[G3] This is an area which requires further investigation.
2.5.3. Short-Lived Fission Gas Release Model

Both the Dragon Project in the United Kingdom, and Pointud and Chenebault in France studied the release rates of short-lived fission gases from particle fuels. They found three mechanisms for the initial release of short-lived fission gases from the kernel. For temperatures less than 600°C, release is driven by gas diffusion. For temperatures in the range from 900°C to 1300°C, bulk diffusion determines release. Finally, for temperatures greater than 1300°C, release is primarily due to recoil and knock out from the kernel directly to the buffer layer.

The steady-state fractional release of gas, they found, depends on temperature and the nuclide half life. The correlations they found have the form of "a power of the nuclide half-life, with the exponent equal to 0.2 ± 0.1" for temperatures in the range of 250°C to 350°C. This exponent approached 0.5 ± 0.1 at temperatures in the range from 850°C to 1300°C.

Studies were performed on failed particles in fuel rods irradiated to low burnups. Particles containing initially dense kernels which were irradiated to low burnups had a release rate to birth rate ratio (R/B) for Kr-85m at 1100°C of 0.005 ± 0.003. When the particles were irradiated to high burnups, the initially dense kernels became porous and the R/B ratio value increased significantly.

They showed experimentally that for porous kernels, there is no dependance of release on burnup, while for initially dense kernels, the release rate is a strong function of burnup. This was attributed to the breakup of dense particles during burnup, which increases the area for release from the kernel. The kernel fabrication route and heat treatment during coating do not seem to influence the diffusion of these gases.

2.6. Predictive Failure Models

The second category of models attempts to predict the behavior of fuels. These models are benchmarked to experimental data. Given the relevant parameters, a model will predict failure rates of fuels as a function of time or temperature.
2.6.1. **Statistical Interpretation of Experimental Results**

England, the Federal Republic of Germany and the United States have built up a large database on coated particle fuel experiments. To interpret this data and determine failure rates, statistical methods must be utilized. A typical experiment will analyze the failure rates of anywhere from 10,000 to 200,000 particles. A large core will contain many orders of magnitude more particles. Many of the determining factors in fuel failure such as the thickness of each coating layer and the actual temperature seen by a particle are statistically distributed, so an absolute failure rate can not be specified. To predict the true failure rate from observed experimental failures requires statistical analysis of the data.

Nabielek has determined a relationship between the observed failure rate, the sample size and the confidence limit set by the person interpreting the data.[N2] When no failures are observed, the relationship is given by

\[ \zeta_{\text{max}} = 1 - \exp\left(\frac{\ln(1-C)}{N + 1}\right) \]

where \( N \) is the number of samples, \( C \) is the confidence level, and \( \zeta_{\text{max}} \) is the maximum failure fraction. For example, if there are no failures observed in a sample of 79,033 particles, \( \zeta_{\text{max}} = 3.8 \times 10^{-5} \) at a confidence level of 95%. Therefore, the particle failure fraction lies somewhere between 0 and \( 3.8 \times 10^{-5} \).

2.6.2. **Kaae Pressure Vessel Failure Model**

The pressure vessel failure model by Kaae [K1] was developed to predict BISO particle failure caused by internal pressures exceeding the fracture stress of the pyrocarbon layers. Kaae's model accounts for factors such as diametral change of the pyrocarbon due to irradiation and creep. The model is based on first principles of mechanics, and is solved iteratively by a computer. The significant thing about Kaae's model is that it was the first pressure vessel failure model to deal with failures statistically. Kaae used a Weibull analysis to calculate the probability of coating failure, where unique failure stress had previously been assumed.
The Kaae model was validated by prediction of BISO dimensional changes. It also allows extrapolation over fluences. Interestingly, Bennett of INEL has revived this approach for predicting fuel behavior in the MHTGR-NPR (New Production Reactor).

2.6.3. **Goodin-Nabielek SiC Model**

The model developed by Goodin and Nabielek is the US-FRG reference model.[G2] To develop a model which would match experimental data on SiC failure, Goodin and Nabielek found that they needed a statistical distribution. Physically, this represented a possible dependance on local SiC density or grain size, or other microscopic parameters which could not be measured. This model gives a failure curve as a function of time or temperature. Their model is based on both the corrosion of SiC by fission products, but also on the decomposition of SiC at high temperatures.

Goodin and Nabielek made a number of assumptions in developing this model. First, they assumed that the thickness of the SiC layers is normally distributed. Next, they assumed that SiC degradation followed a log-normal distribution based on the observation that “many core variables have uncertainties that are skewed to higher values.” Finally, they said that the SiC degradation rate “has an Arrhenius-type temperature dependance.” These assumptions are mathematically approximated by a Weibull distribution of the form:

$$\Phi = 1 - \exp[-\ln2^*(kt)^m] = 1 - 2^{-(kt)^m}$$

where

- $\Phi$ = SiC failure fraction
- $k$ = decomposition or corrosion frequency factor ($h^{-1}$) (given below)
- $t$ = heating time ($h$)
- $m$ = Weibull parameter (discussed below)

$$k = k_0 \exp \left[ -\frac{Q}{RT} \right]$$

where

- $k$ = frequency factor ($h^{-1}$)
- $k_0$ = constant
- $Q$ = activation energy ($J/mol$)
- $T$ = temperature ($^\circ K$)
\[ R = \text{gas constant} \ (8.314 \ J / \text{mol} \cdot ^\circ\text{K}) \]

\( m \) can be related to \( k \) based experimentally on the thickness distribution of the SiC layer.

In order to combine failures due to corrosion and decomposition, Goodin and Nabielek use the following relationship:

\[ \Phi_T = \Phi_C + \Phi_D \ (1 - \Phi_C) \]

where

\[ \Phi_T = \text{total failure fraction} \]
\[ \Phi_C = \text{failure fraction attributed to corrosion} \]
\[ \Phi_D = \text{failure fraction attributed to decomposition} \]

Once the SiC layer fails, the outer pyrocarbon layer remains as an additional diffusion barrier. Experiments show that diffusion rates through the outer pyrocarbon layer do not depend on prior irradiation history. This is reasonable since these rates were measured after the SiC failed, and by that time, the temperature had been high enough for long enough to anneal out any radiation defects in the pyrocarbon. The Goodin-Nabielek does not account for any significant hold up of fission products in the outer pyrocarbon, which is the distinctive feature of the next model.

2.6.4. **Martin-Goodin-Nabielek Failure Model**

The Martin-Goodin-Nabielek model (MGN) is designed to predict cesium and krypton release from particles or pebbles.[M1] It is based on a “statistical distribution of failure times followed by a diffusive release from the failed silicon carbide.” A Weibull distribution essentially the same as used by the Goodin-Nabielek model is the basis for this model. This model is an attempt to update the Goodin-Nabielek model after more data became available, particularly data on the distribution of cesium in the matrix material. At Seibersdorf and Harwell, the fractional release from a particle was plotted versus the fraction of the total population of particles with that fractional release. This plot shows a sharp peak for very low releases. Ten percent of the particles, however, had fractional
releases from 20 - 100%. This spectrum fits neither a pure diffusion model, nor a pure pressure vessel failure model.

The MGN model is phenomenological. It is based on physical principles and measured values for constants whenever possible. Martin claims that the degree to which their model agrees with experimental data is a measure of how well they understand the physical processes involved. For this model, silicon carbide failure is defined as permeability of the silicon carbide to cesium.

New data shows that some cesium which is released from the particles is retained in the matrix of irradiated pebbles. An extreme example of this is the FRJ2-K13/4 test, in which 96% of the cesium released from the particles remained in the matrix. Early models assumed that there would be no retention in the matrix. When analyzing the data with this assumption, the number of failed particles would be significantly underestimated.

To deal effectively with this new cesium data, the MGN model makes two major assumptions. The first is that diffusion of cesium in the bulk of the SiC grain is slow enough to be neglected in the temperature range of interest. Then second is the once grain boundary damage reaches the stage where the cesium diffusion coefficient begins to rise, the grain boundary damage proceeds on a time-scale short compared to the time scale for the diffusion. Diffusion through intact SiC is approximated as zero. Once the SiC has failed, it is assumed that the diffusion coefficient is measurable experimentally. A statistical correlation is used to determine the statistical distribution of failure times. The MGN model has been applied to tests in the $1700 - 2000^\circ\text{C}$ range with some success. Further development of this model will depend on measurements of the distribution of releases from individual particles.

2.7. Future Directions for Fuel Development

Predictive fuel models, such as those just described, will allow engineers to forecast the behavior of coated particle fuel under many sets of conditions. These models provide a
method for predicting fuel behavior, even when the conditions of interest are not exactly represented by the available experimental results. In order for predictive fuel models to be fully developed, the models must be based on the failure phenomena, in order for fuel failure to be accurately represented. In order for phenomenological models to be possible, the failure mechanisms themselves must be understood. The fuel test facility will provide a means of investigating the fuel failure mechanisms.

Improvements to TRISO fuel may be possible, allowing an HTGR to run hotter, or providing more of a safety margin. Among the potential improvements to fuel are altering the coating thicknesses and using different coatings, such as zirconium carbide. There is currently not a strong push for fuel investigation by commercial producers, but the fuel test facility will provide a means for academic research into fuel failure phenomena as well as ways to improve fission product retention.
Chapter 3

3. System Overview

3.1. Design Goals

The purpose of the coated particle fuel test facility is to give researchers at MIT a tool to do coated particle fuel studies. Our primary goal is to provide an irradiation environment similar to what fuel experiences in a high temperature gas reactor. First, we must set a goal for the temperature of the fuel sample. Our design temperature for the sample is 1000°C. This is somewhat higher than expected operating temperatures and so will allow research into the effects of running reactors hotter than they are currently designed for.[M2] During accidents, fuel temperatures may reach 1600°C after many hours. The fission chain reaction will stop long before fuel reaches such high temperatures. Therefore, it is only necessary to irradiate fuel at operating temperatures. If desired, the fuel samples can be heated after they are removed from the irradiation facility, with no loss in accurate reproduction of accident conditions. Our goal is to be able to maintain the fuel sample’s temperature regardless of the operating power of MITR-II. Minimizing temperature swings in the sample eliminates a source of uncertainty in our experiment.

In addition to temperature, the atmosphere of the experiment is an important part of the environment which must be reproduced. Current designs for high-temperature gas-cooled reactors use helium as the coolant. With the goal of reproducing the environment for irradiation, we will want to operate our facility with an inert helium environment. Since helium in a commercial reactor is not completely pure, we would also like to be able to investigate if the impurities in reactor coolant affect fuel performance.
With the goal of accurately reproducing the environment of a high temperature gas reactor comes the goal of reproducing the neutron energy spectrum and flux. Among different gas cooled reactors, there is a wide variation in the energy spectra. Typical fluxes in AVR, a German pebble bed reactor, shown in Table 3.1.[Z3] If higher fluxes could be achieved in the test facility, less time would be required to reach the desired burnup. Predicted fuel burnup for the German HTR design is 8.93% FIMA.[N3] We want to be able to achieve fuel burnup in our facility of at least 8.93% FIMA. The HTR design also states that the expected fast fluence when particles reach a burn-up of 8.93% FIMA is $2.1 \times 10^{21}$ neutrons per square centimeter, measured for neutron energies greater than 0.1 MeV.[N3] We would like to be able to irradiate particles to the predicted burn-ups, and match the fast fluence as closely as possible.

<table>
<thead>
<tr>
<th>Energy Range</th>
<th>Neutron Flux (neuts/cm² sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$E \leq 1.9$ eV</td>
<td>$5.108 \times 10^{13}$</td>
</tr>
<tr>
<td>$E &gt; 1.9$ eV</td>
<td>$5.332 \times 10^{13}$</td>
</tr>
<tr>
<td>$E &gt; 0.1$ MeV</td>
<td>$1.896 \times 10^{13}$</td>
</tr>
</tbody>
</table>

### 3.2. Design Constraints

There are a number of constraints placed on the design. First and foremost is safety. This facility must meet all safety requirements of the MIT reactor. Operating this facility will involve working with high temperatures, radioactive samples, and high radiation fields in the experiment. Thermal and radiation shielding are very important. A highly reliable design will minimize maintenance, reducing the doses workers will receive. The simpler the facility is to install and operate, the less time workers will be exposed to radiation. Placing this facility in the MIT reactor adds many additional constraints. Space in the reactor is limited, and we must be able to design our facility to be small enough to fit in
whatever space is available, including limiting our samples to be relatively small. Space for equipment is also limited. Any equipment which is not very small must go in the equipment room below the reactor. The equipment room is very crowded, and there is no room for large pieces of equipment.

The reactor's license is very specific about what experiments can be run, and what requirements they must fulfill. These license requirements are most restrictive on experiments which will go directly into the core, and they are somewhat less restrictive on experiments which will be placed in ports around the core. In addition to designing the fuel test facility to meet the license requirements for the MIT reactor, the facility will have to meet other regulatory requirements. Special permits are required for fuel possession. Keeping the amount of fuel small may make acquiring the necessary permits simpler. It will certainly minimize disposal costs.

Our goal temperature for the fuel of 1000°C is much higher than any temperatures in the MIT reactor. There are temperature alarms in the D₂O tank set to 55°C, and the D₂O temperature is 45°C when it returns from the heat exchangers.[K3] The fuel test facility can not disturb the thermal condition of the reactor significantly. The heat exchangers for the reactor coolant and for the D₂O reflector have an extra kilowatt of heat removal capacity, so the fuel test facility can not add more than 1 kW to the D₂O.

In summary, the design goals and the constraints on the proposed facility are

- Neutron flux as high as possible
- Fuel Temperature up to 1000°C
- Fuel temperature variation during irradiation < 10°C
- Thermal gradients in fuel sample <50°C/cm
- Sample space at least 1 cm diameter, 3 cm long
- Maximum D₂O temperature less than 55°C
- Total heat rejection to D₂O < 1kW
- Fuel burnups to 20% FIMA or greater
3.3. The TRIGA King Furnace Facility

A similar facility, the TRIGA King Furnace Facility, was used in a General Atomic’s TRIGA Mark I reactor in the late 1960’s.[A1] The TRIGA King Furnace Facility (TKFF) was designed for use in the core of a TRIGA research reactor. The facility was designed to be inserted in place of a fuel element in the TRIGA reactor core. The furnace of the TKFF was a graphite heating element 8 inches long, with a 1 inch OD and a 0.75 inch ID. Two molybdenum radiation shields provide thermal insulation for the furnace. The containment vessel for the furnace was made of aluminum, and sized to fit in the place of a fuel element. The containment vessel serves as one of the electrical leads for the furnace. The other lead is a second aluminum cylinder concentric within the containment vessel, which connects to the top of the furnace. The inner aluminum pipe is connected to the upper flange of the facility, which has a window for measuring the temperature of the sample using optical pyrometry. By disassembling the upper flange, the sample can be lowered into the facility using a modified fishing reel and line. The sample can be removed in the same way.

Figure 3.1 shows the furnace section of the TKFF. The TKFF was designed to be operated either with a static inert gas pressure in the system, or with continuous purging of the gas. The purging system included a filtration system, so fission gas releases could be monitored.

The TKFF was used for a variety of experiments. Experiments studying the temperature dependance of fission gas release from coated particles were performed. Iodine release experiments were reported as well. Another type of experiment which was run in the TKFF was transient studies. These studies relied on the pulsing feature of TRIGA reactors to provide a burst of power to the particles. The temperature of the particles was tracked as a function of time and compared to the reactor power as a function of time.
The TRIGA King Furnace Facility provided an experimental facility which meets many of the goals we have set for our fuel test facility. In some important respects, however, it differs. These are mainly due to the differences between a TRIGA Mark I reactor and the MITR-II. The TRIGA King Furnace provided the inspiration for using electrical resistance heating, using the piping as part of the electrical circuit, and using optical pyrometry to measure the sample temperature accurately.

3.4. Interfacing with MITR-II

The MIT research reactor, MITR-II, is a light-water cooled, heavy-water reflected reactor. The reactor uses highly enriched fuel in aluminum plates. The core is in the shape of a hexagon with the fueled region having the following dimensions: 15 inches across the flats (0.381 m) and 22.375 inches tall (0.568 m).[M4] The coolant and core sit within the core tank, which is about 22 inches (0.559m) in diameter. The reactor has a maximum power of 5 MWth. It is typically operated continuously from Monday morning until Friday afternoon, when it is shutdown for the weekends.

Since it was designed as a research reactor, MITR-II has an ample supply of experimental facilities. The reactor has three dummy elements in the core which can be replaced with experiments. There are beam tubes which provide beams of neutrons from under the reactor core. There are also ports which go into the D$_2$O reflector, or the graphite surrounding the reflector. The ports which go into the reflector are all horizontal, but there are some vertical and some horizontal ports in the graphite region.
Figure 3.1
TRIGA King Furnace[A1]
Two potential locations for the facility in the MIT reactor were identified. Both provide large neutron fluxes and relatively fast neutron spectra. The first option is to place the facility directly in the core in a dummy element. The second option is to put the facility in a horizontal beam port in the D$_2$O reflector, designated 6RH2. The tip of this port is only a few centimeters from the core tank, and it is at the height of the core mid-plane. In the opinion of Kwan Kwok, the superintendent of reactor operations at the MIT reactor, if the facility were located in the core, safety issues such as the floodable volume and the neutronic effects of the experiment on the core performance may be prohibitive. Placing the facility in the beam port, however, would minimize the impact on the reactor operation, and would not pose as difficult a safety question. Therefore, the facility was designed for insertion in the 6RH2 beam port.

A cross-section of the 6RH2 port is shown in Figure 3.2. The tip of the port which projects into the heavy water reflector tank has an inner diameter of 3 inches (76.2 mm). This will set the size scale of the facility. The length from the outer wall of the reactor shielding wall to the facility tip is 9 feet, 4 inches (2.84 m).[M4] These dimensions will lead to a long thin facility, with the irradiation chamber near the tip of the port.
Figure 3.2
6RH2 port cross-section
3.5. Sample Design and Manufacture

As stated in the design goals, this facility will be used to irradiate small fuel samples. The particles will either be mixed with a binder and placed inside a graphite holder, or placed loose in a holder with a plug to hold them in. In order to determine the volume of the fueled region of the sample, we must make some assumptions about how the particles will pack. Let \( d \) be the diameter of a particle, and \( D \) be the diameter of a cylinder into which the particles will be packed. Then, if \( D = 4 \, d \), approximately 8 particles will fit in a flat layer, allowing some room for binder between particles. (Since the particles will be surrounded on all sides by a graphite holder, the binder is not structurally significant, and so there is never a need for the binder to be structurally significant.) If the layers spaced 1 \( d \) from center to center, then we can put 24 particles in three layers, or 3 \( d \) total height. Thus, for 24 particles in one sample, the fueled region can be a cylinder 4 \( d \) in diameter and 3 \( d \) tall. The diameter of coated particles now in use in Germany is approximately 900 \( \mu m \). If we assume a diameter of 1 \( mm \) for particles, then our fueled region will have a 4 mm diameter and a height of 3 mm. An inter-layer spacing of 1 \( d \) does not account for the fact that spheres will actually pack closer together than that due to shifting of one layer onto another. The estimate for the size of the fueled region is not intended to be an actual calculation to determine the packing density, but just to determine the general size of the region.

To determine our overall sample diameter, we must allow for the structural graphite region outside the fueled region, as well as allow enough clearance for insertion of the sample into the heater. If we make the outer diameter of the sample 9 \( mm \), that will provide just over 2 \( mm \) of graphite around the fueled region. Two millimeters of graphite surrounding the fueled region will provide sufficient structural support, while an outer radius of 4.5 \( mm \) will fit in the heater (with a 6 \( mm \) inner radius) easily.
Since the heater will have a large current flowing through it, the sample must be electrically insulated. Otherwise, the electric current would flow through the sample, increasing the cross-sectional area for current flow and decreasing the ohmic heating rate in that region. Machining an insulating cylinder of a ceramic insulator is possible. There are materials which are appropriate for the temperatures in the facility and are electrically insulating. However, the insulating cylinder would be very thin-walled and therefore very difficult to make. In order to circumvent machining problems with the insulator, we turned to insulators which are liquid when applied. Cements for binding insulation are a good option. By using a liquid, the sample can be made, and painted with the cement. The cement layer can be as thin as 0.254 mm, although information on the electrical resistivity will be necessary to determine if that is a sufficient thickness. Zircar® markets an alumina cement which can be used to temperatures of 1650°C.

Twenty-four particles per sample is a small quantity for commercial fuel producers to manufacture. This leads us to consider setting up a desktop fuel production facility. The coatings for particle fuel can be made in a desktop-sized facility. Manufacturing the kernels, however, would be a more difficult operation, and it may be easier to purchase the kernels from a fuel manufacturer. A desktop production facility would allow researchers to adjust the fuel composition for one experiment as the results from another experiment are analyzed. Researchers would be able to run an experiment, analyze the results, and change coating parameters, such as coating thickness, or coating materials. Greater experimental flexibility would be gained by having a fuel production facility at MIT.

### 3.6. Heat Source

One of the variables we must be able to control is the temperature of the sample. The operating temperature range of fuel in the HTR-MODUL, a German high temperature gas reactor, is from 700 to 900 °C.[N3] For safety, the fuel test facility will be designed to have a maximum temperature 500°C when there is no electric heating being supplied. This
requirement will assure that if power is lost, the facility will not overheat, and the structural integrity of the facility will not be challenged due to excessively high temperatures.

When the reactor is operating at its full power of 5 MWth, and no electric power is being supplied, the maximum temperature in the fuel is about 273°C. With MITR-II at full power, only 250 watts of electric power is needed to keep the fuel at 1000°C. The maximum heater power the facility will need is that power which is necessary to maintain a fuel temperature of 1000°C when the reactor is shutdown and there is no heating due to neutrons and gamma rays. This maximum heater power was calculated to be 300 watts. (Calculations for these heating rates will be discussed in Chapter 5.)

3.7. Control and measurement systems

There are two variables which will be measured. The first is the sample temperature and the second is the gas composition. Knowing the sample temperature accurately is important to the analysis of the fuel performance. Since the sample will be at temperatures in the range 600–800°C, optical pyrometry was chosen to measure the sample temperature. The geometry of the facility, a very small sample at the end of a long, narrow pipe, makes the optics of the pyrometry system very important. One company, Mikron®, has developed pyrometry systems for facilities with high temperatures and high radiation fields. Mikron® developed the equipment used for temperature measurement in the TRIGA King Furnace Facility. A control loop can be set up to adjust the heater power based on the measurement of the sample temperature.

The gas composition in the atmosphere is important is so far as it must be sufficiently pure that the heater, when at high temperatures, does not react chemically with any impurities in the atmosphere. Since the fuel will be inside a graphite holder, it is unlikely that contaminants in the gas will diffuse through the carbon holder without reacting first with the holder. For that reason, the purity of the atmosphere should not effect the fuel performance.
3.8. Sample Handling

Due to the location of the port, there is a very high gamma dose at the end of the port when the shield plug is removed. This makes the insertion and removal of the samples difficult, since insertion and removal must be performed from behind radiation shielding. By making the inner diameter of the facility constant along its length, it is possible to insert the sample simply by pushing it in. To remove the sample, there must be a way to hold on to the sample until it is in a shielding container, and then to remove the tool, leaving the sample in the shielding container. A tool has been developed which will fit into an opening in the sample and grip the sample. The sample can then be pulled into a shielded container. Once the sample is in the container, the tip of the tool, the part gripping the sample, is removed from the pole which had been used to insert it to the end of the facility to grab the fuel. The tool end will provide a means to hold on to the sample after irradiation, while simplifying the removal of the sample from the facility to a shielded container.
Chapter 4

4. Neutronics

Neutronics issues associated with the fuel test facility can be grouped in two major categories: effects of neutron spectra and fluxes on the sample, and heating due to neutrons and gamma rays from the core. The 6RH2 port has not been used in approximately 20 years, and no accurate data are available on neutron spectra, fluxes, or heating rates in the port. Since experiments to measure the heating rate were not covered in the scope of this project, other sources were investigated. 6RH2 is one of a pair of ports which mirror each other from opposite sides of the thermal column in the MIT reactor. The other port, 6RH1, is used for the pneumatic rabbit tube, where neutron activation is performed for neutron activation analysis. Unfortunately, there is no data on the precise flux levels, or the spectrum, in the neutron activation analysis facility. Also, the residence time of neutron activation samples in the region of high flux is very short and the sample masses are small, so no data is available on heating due to gamma rays and neutrons.

Since no experimental data is available for 6RH2, analytical information was sought. A numerical simulation of the MIT reactor was developed by Everett Redmond, a graduate student in the nuclear engineering department. The model, which includes the core, D\textsubscript{2}O reflector, graphite region surrounding the reflector, and the medical facility beneath the reactor was developed for the code “Monte Carlo Neutron and Photon Transport Code System” (M5).
A model of the fuel test facility was developed and incorporated into the reactor model. From this model, we were able to calculate neutron and gamma heating rates, as well as the neutron flux and spectrum. To verify the accuracy of the model, an experiment to determine the heating rate due to neutrons and gamma rays was also modeled. The experiment, performed by Steve Boeright, used an ionization chamber lowered into a dummy fuel element to estimate the heating.[B3] The heating rates calculated by MCNP for Boeright’s experiment did not agree with Boeright’s data, but a comparison of the predicted and measured heating rates enabled us to make a conservative normalization of the MCNP data.

4.1. Description of MCNP

MCNP is a time-dependant, continuous-energy, coupled neutron / photon Monte Carlo transport code.[M5] MCNP can access over 500 neutron and photon interaction tables. The information from these interaction tables is used when MCNP tracks particles through the input problem. MCNP requires the user to define the geometry of their system explicitly using geometry primitives.

4.2. The MCNP Model of the MITR-II

The basic model of the MIT reactor built by Everett Redmond includes everything inside the concrete shielding. The core of the reactor is modeled explicitly, including the fuel and clad in individual fuel plates. The core tank, with the core and light water inside it, are included, as is the reflector tank, filled with D$_2$O. The model extends two feet above the top of the core, and down to include the medical therapy room below the reactor. An axial view of the reactor as it is modeled is shown in Figure 4.1. The core model used in the calculations for the fuel test facility was the core which was in the reactor in early December, 1990, and the control blades were modeled in the position they were recorded to have been in at the time of start up on December 4, 1990. There were
two experiments in the core for this model. One was the PCCL loop, and the other is a gamma heating experiment which was performed by Steve Boerighter. Both of these experiments were in the center ring of fuel elements— the PCCL loop was in the A1 position, and the gamma heating experiment was in the A3 position. All other locations contained fuel elements.

A model of the experiment performed by Steve Boerighter was included to validate the model. The experiment was placed inside a dummy element in the center ring of fuel in the reactor. The dummy element, which has the same rhomboidal shape as a fuel element, is made of aluminum-6061 with an elliptical hole in the center. The thimble for the BCCL experiment, which is an elliptical pipe, was inserted into the dummy element. An aluminum thin-walled 1 inch diameter tube was lowered into BCCL thimble. The space between the pipe and the thimble was filled with lead shot to decrease the floodable volume. A 3/4” - 0.035 wall aluminum-6061 tube was placed inside the 1” aluminum tube, and an ionization chamber was lowered into this tube. Measurements of ionization were made at 2 inch (5.08 cm) intervals from the bottom of the aluminum pipe to about 6 inches (15.24 cm) above the fuel. The ionization chamber had been calibrated using a known source, and a conversion from ionization to heating in watts / gram water was made.

Another experiment performed by Zaker in 1977 was based on calorimetric techniques.[Z1] Zaker measured heating in carbon, aluminum, and beryllium. Time did not permit modelling of Zaker’s experiment using MCNP. It is informative to note, though, that the results of Boerighter’s experiment agree closely with Zaker’s data over much of the core region. (Zaker’s data does not include any measurements above the core.) Figure 4.2 shows the heating estimates Boerighter’s experiment as well as heating measurements done by Zaker.
The model of Boerighter's experiment used in MCNP is shown in Figure 4.3. The geometry of the experiment was modeled as closely as possible. The lead shot, however, had to be modeled as homogeneous lead, with a correspondingly lower density. Unfortunately, there is no information available on the packing density of the lead in the experiment. To approximate this, the region between the elliptical hole and the pipe Boerighter used was cut into 4 regions as shown is Figure 4.3. The two regions which lie between the ellipse and the pipe on the ellipse's minor axis are assumed to have no lead in them. Since there is little room, it is not likely that any shot would have fit in these regions. In the two regions between the pipe and the ellipse along the ellipse's major axis, the density corresponding to randomly packed lead shot was used. To determine the random packing of lead shot, a circle was fit to the regions of interest, with a diameter of approximately 2 cm, and a lead shot diameter of 1 cm was assumed. The packing density as a function of the ratio of outer diameter to shot diameter is known, and for the dimensions we have assumed, the packing fraction is 0.55.[E1] This leads to using a density of lead equal to 6.3 grams / cm³.

A model of the fuel test facility was developed and inserted into the 6RH2 position which was already included in the model of the reactor. Figure 4.4 shows the facility in the port, relative to the core location, at the height of the centerline of the 6RH2 port, approximately 6 inches above the bottom of the core. A detailed diagram of the fuel test facility is shown in Figure 4.5. As long as there are no materials which are strong absorbers of neutrons or gamma rays, the exact materials and geometry of the facility will not strongly affect the results. Since our facility will consist mainly of aluminum, graphite, silicon and oxygen, which all have small absorption cross-sections, the precise geometry within the facility will not have a strong impact on the results.
Figure 4.1
Axial view of MITR-II as modelled in MCNP.
The errors on Zaker's data are smaller than the symbols.

Figure 4.2
Heating estimates by Boeriger and Zaker.
Figure 4.3
Model of Boerighter's experiment for MCNP.
Figure 4.4
Cross-sectional view of the MCNP model through the core and the 6RH2 port.
Figure 4.5
Cross-section of the fuel test facility as modelled in MCNP.

The model of the facility ends at a plane perpendicular to the port four inches into the graphite. Heating calculations and neutron flux levels will be most interesting in the section of the facility which extends into the D$_2$O. For this reason, the model extended beyond the D$_2$O tank. Beyond the plane ending the model of the facility, it is assumed
that the heating rates are very small, an assumption supported by the results of the calculations.

The geometry and materials used for the facility correspond to the heater design described in Chapter 5, with a "fin" attached to the heater. Ultimately, this geometry was not chosen for the tip of the facility, but as discussed above, the error in using data from this model for the actual design is small.

4.3. **MCNP Results and Analysis**

Figure 4.6 shows the heating rate calculated by Boerighter as well as the heating rate calculated by MCNP, both normalized to a reactor power of 5MWth. There is a difference over most of the region of about a factor of 3. There are a number of sources of uncertainty in the MCNP model, as well as some issues not addressed by Boerighter. Each of these will most likely have some effect on the results, and combined, they may explain the difference between the MCNP predictions and Boerighter's calculations. To begin with, the lead shot density was not known. The diameter of the shot used was not recorded, nor was the height to which the shot was used. The conversion of ionization rates to heating rates may contain approximations which will introduce some error. In the MCNP model, delayed gammas from the decay of fission products were not included in calculations, and neither were delayed gammas from aluminum activation. MCNP does not have a way to model an ionization chamber. Since Boerighter's data was converted to heating per gram of water, the ionization chamber was replaced by a volume of water in MCNP, in which heating was calculated.

MCNP results and Boerighter's data agree most closely for those points which are above the fueled region of the core. This suggests that some of the difference between the calculations may disappear as one looks further from the fueled region. Since the fuel test facility will not be in the core, only those points above the fueled core will be
considered to determine the accuracy of the MCNP data for the facility. Comparing heating rates at these points, we find that the ratio of Boerighter’s calculated heating rates to MCNP’s heating rates is approximately 3. To insure that the neutron and gamma heating rates used are conservative, MCNP heating rates will be normalized to Boerighter’s data by this factor of 3.

To calculate the heating rates using MCNP, the fuel test facility was broken into regions both axially and radially. Combined neutron and gamma heating rates were reported by MCNP, along with a measure of their error. To plot the heating rates as a function of position, the heating rate was plotted versus the mean location of the computing region. Figure 4.7 shows the heating rate as a function of distance from the tip of the port nearest the reactor core, calculated for the insulation between the facility and the port wall. Both the MCNP calculation, and the MCNP calculation corrected by normalizing to Boerighter’s data are shown. To study the heating rate as a function of radial location in the port, the axial region including the heater was used. The radial heating profile is shown in Figure 4.8, without the normalizing factor from Boerighter’s data. It can be seen in Figure 4.8 that there is not much information available from MCNP for this. The error bars on the data are quite large, and the spread is also large. For this reason, and again, to be conservative, the heating rates will be assumed to be constant over the radius of the port, and the heating rate used will be the heating rate calculated for the insulation between the port wall and the test facility’s outer pipe (the region used for the axial profile).
Figure 4.6
Heating rates as calculated by Boerighter and as predicted by MCNP.
Figure 4.7
Axial heating profile in the Fuel Test Facility, both uncorrected and corrected for Boerighter’s data.

Equation for curve fit to corrected MCNP data:

\[ y = 0.41285 \times e^{(-0.051125x)} \quad R = 0.99242 \]
Figure 4.8
Radial profile of heating by neutrons and gamma rays in the heater section.
To use the heating data as a function of position along the port, an exponential curve 
was fit to the corrected heating rates in Figure 4.7. The equation of the curve is
\[ y = 0.41285 \exp(-0.051125 \times) \]
where \( y \) is the heating in watts / gram, and \( x \) is the distance from the tip of the port, in cm.

In addition to using MCNP to calculate the heating rates in the fuel test facility, 
tallies were also done on the neutron flux at the center of the heater, as well as the 
neutron current crossing the plane ending the heater. The flux in the center of the heater 
was calculated for three energy groups. The fluxes are presented in Table 4.1. The net 
neutron current across the plane ending the heater is \( 2.66 \times 10^{14} \) neutrons / sec. This is the 
current across the entire cross-section of the port when the reactor is at full power. This 
neutron current can be used to estimate the neutron flux which will have to be shielded at 
the end of the facility. No tally was done for gamma rays crossing the plane.
Unfortunately, the MCNP model does not account for delayed gammas from fission 
products, so any gamma calculation would have been an underestimate.

<table>
<thead>
<tr>
<th>Energy (MeV)</th>
<th>Flux in AVR (neutrons / cm² sec)</th>
<th>Predicted flux in FTF (neutrons / cm² sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td>( \geq 1.9 ) eV</td>
<td>( 5.108 \times 10^{13} )</td>
<td>( 5.67 \times 10^{13} )</td>
</tr>
<tr>
<td>( \geq 1.9 ) eV</td>
<td>( 5.332 \times 10^{13} )</td>
<td>( 7.11 \times 10^{12} )</td>
</tr>
<tr>
<td>( \geq 0.1 ) MeV</td>
<td>( 1.896 \times 10^{13} )</td>
<td>( 3.16 \times 10^{12} )</td>
</tr>
<tr>
<td>Total</td>
<td>( 1.044 \times 10^{14} )</td>
<td>( 6.381 \times 10^{13} )</td>
</tr>
</tbody>
</table>

From the predicted fluxes in the center of the heater, we can estimate the fission rate in the particle, and estimate how long it will take to reach desired burn-up levels. 
Appendix A contains the calculations to estimate the burn-up time per percent FIMA. A 
rough estimate is that it will take about 37 days of irradiation with the reactor at full 
power per percent FIMA. To reach 20% FIMA, then, will take 743 full power days. This
is longer than desired, as we would like to be able to turn samples around faster than this. We could look for a higher flux inside the core, but the safety questions which lead to the choice of 6RH2 tend to discourage that. Given that irradiation to 20% FIMA will take so long, it may be reasonable to focus on lower burn-up irradiations initially. Higher enrichments for the particles is another potential way to increase the fission products in the particle, which will allow investigation of phenomena of fission product attack of the particle coatings.

To reach a burn-up of 8.93% FIMA, the HTR design point, it will take 332 full power days. Then, since we know the neutron flux with energies greater than 0.1 MeV, we can calculate that the fast fluence would be $9.06 \times 10^{19}$ neutrons / cm$^2$ when the fuel reaches 8.93% FIMA. The anticipated fast fluence in HTR corresponding to a burn-up of 8.93% FIMA is $2.1 \times 10^{21}$ neutrons / cm$^2$. Our predicted fast fluence is a factor of 20 lower than that predicted for HTR. This is lower than desired, but there does not appear to be a simple solution.
Chapter 5

5. Thermal Design

There were two major influences on the thermal design of this facility. The first is our goal to be able to maintain the temperature of the samples at temperatures up to 1000°C for extended times, regardless of the reactor’s operating power. The second influence was the effect the facility would have on MITR II. The heater and sample section of the facility will be near the tip of the 6RH2 port. This end of the port is in the D2O tank. The D2O has a heat exchanger to cool it, and the circulating D2O is kept at an average temperature of 45°C. The D2O heat exchanger can handle up to an additional kilowatt of heat removal. There are temperature alarms in the D2O set at 50°C, and for neutronic stability, boiling in the D2O is not allowed. A final consideration was the provision for safe shutdown of the facility if external power to the facility’s equipment is lost.

This last point leads to a design which requires active heating, and which, in the case of equipment failure, would slowly cool down by conducting heat to the D2O. To minimize the thermal effects on the D2O as well as to minimize the electric power requirement, the obvious approach is to insulate the facility very well. However, if there is too much insulation, neutron and gamma heating from the reactor may be enough to maintain the temperature of the sample above the desired operating point, so a balance must be achieved on the level of insulation.

Since our heater will be at temperatures up to 1000°C, any insulation used must be rated for use at these temperatures. Ceramic fiber insulations are available with operating
temperatures greater than any expected in the facility. At the temperatures expected for the heater, radiative heat transfer can be quite substantial. Both radiation shields and fibrous insulation were investigated.

5.1. Heater configuration

The heater is a thin-walled electrically heated graphite cylinder. Electric current flows along the outer pipe of the facility (not the port wall), in via the bottom disc to the heater base, through the heater and back along the inner pipe of the facility, as shown schematically in Figure 5.1. The heater section has been designed with a much higher electrical resistance per unit length than any other part of the circuit. This was accomplished by minimizing the cross-sectional area of the heater, and by choosing relatively low resistance materials for the remainder of the circuit. This will concentrate the heating at the desired location. Table 5.1 shows electrical resistances per unit length for different materials and configurations. The configurations described in the table will be discussed in greater detail later in this chapter.

The heater must be long enough that the thermal gradient experienced by the sample is small. Thermal gradients can affect coated particle fuel failure. This effect already been studied, and removing the influence of thermal gradients will simplify experimental analysis later.[W1] In AVR, a German high temperature gas reactor, the fuel typically experienced a thermal gradient less than 30°C/m.[E2] This is the maximum thermal gradient we are willing to tolerate in our samples.

A graphite heater is being used for a number of reasons. Graphite retains its structural properties at the elevated temperatures we need for the heater and it will not react with the sample. In addition, graphite has favorable neutronic characteristics. It will not significantly attenuate the flux before it gets to the sample, its properties are known as a function of fluence, and it will not become too activated, an important consideration for
maintenance and disposal. As the total neutron dose to the graphite increases, the electrical resistivity increases, with the result that power will be better concentrated in the heater section, and lower currents will be required for the same power. Additional information on the selection of graphite as the heater material will be presented in chapter 6.

**Table 5.1**

Electrical resistance per unit length in different parts of the electrical heating circuit.

<table>
<thead>
<tr>
<th>Region</th>
<th>Resistivity (Ohm-m)</th>
<th>Cross-Section (m²)</th>
<th>Resistance per m (Ohm/m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heater (OD 7 mm, ID 6 mm)</td>
<td>7.366x10⁻⁶</td>
<td>2.8x10⁻⁵</td>
<td>2.631x10⁻¹</td>
</tr>
<tr>
<td>&quot;Fin&quot; (graphite, OD 28mm, ID 23mm)</td>
<td>7.366x10⁻⁶</td>
<td>2.55x10⁻⁴</td>
<td>2.889x10⁻²</td>
</tr>
<tr>
<td>Extension (graphite, OD 11mm, ID 6mm)</td>
<td>7.366x10⁻⁶</td>
<td>8.5x10⁻⁵</td>
<td>8.666x10⁻²</td>
</tr>
<tr>
<td>Outer pipe (graphite, OD 35mm, ID 31mm)</td>
<td>7.366x10⁻⁶</td>
<td>2.64x10⁻⁴</td>
<td>2.79x10⁻²</td>
</tr>
<tr>
<td>Outer pipe (AL-6061, OD 35mm, ID 31mm)</td>
<td>40x10⁻⁹</td>
<td>2.64x10⁻⁴</td>
<td>1.515x10⁻⁴</td>
</tr>
<tr>
<td>Al-6061 connected to fin (same OD, ID as fin)</td>
<td>40x10⁻⁹</td>
<td>2.55x10⁻⁴</td>
<td>1.569x10⁻⁴</td>
</tr>
<tr>
<td>Al-6061 connected to extension (same OD, ID as extension)</td>
<td>40x10⁻⁹</td>
<td>8.5x10⁻⁵</td>
<td>4.706x10⁻⁴</td>
</tr>
<tr>
<td>Heater base (solid graphite, OD 7mm)</td>
<td>7.366x10⁻⁶</td>
<td>1.539x10⁻⁴</td>
<td>4.785x10⁻²</td>
</tr>
<tr>
<td>Disk (at r = 7mm, 3 mm thick)</td>
<td>7.366x10⁻⁶</td>
<td>1.32x10⁻⁴</td>
<td>5.58x10⁻²</td>
</tr>
<tr>
<td>Disk (at r = 31mm, 3 mm thick)</td>
<td>7.366x10⁻⁶</td>
<td>5.84x10⁻⁴</td>
<td>1.261x10⁻²</td>
</tr>
</tbody>
</table>
Figure 5.1
Electric circuit of heater, schematic
5.2. Transitions from high temperature to low temperature

The heater section, for reasons discussed above, will be made of graphite. The rest of
the electrical circuit, however, does not have the same requirements that led to the choice of
graphite for the heater. In fact, a material with a lower electrical resistance is preferred.
From the heater, the facility will extend approximately 2 meters away from the reactor’s
core. Using a metal for most of the facility’s length would simplify the manufacture, and
improve the structural performance of the facility. Al-6061 is the material chosen, largely
for its superior neutronic characteristics: low neutron attenuation, and minor activation.

Al-6061 does have one major drawback. It’s melting point is in the range 582-652°C.
We have chosen to limit the aluminum’s temperature to a maximum of 300°C. To
electrically connect the graphite to the Al-6061, we must find a way to lower the
temperature of the graphite before the connection is made. This is done by providing a
section at each end of the heater with a large enough cross-section that electrical heating is
small, and extending this section far enough that the temperature of the graphite at the ends
of these extensions is below our maximum temperature limit on the Al-6061.

At the end of the heater which connects to the inner pipe, two design options were
considered. The first option was a “fin” section made of graphite, so-called because of
small cross-section relative to its length. The “fin” is a length of graphite which flares out
to decrease heat generated due to the electric current. The “fin” provides a section of
graphite in which the electrical heating is relatively small, so that the end of the fin is at a
low enough temperature that it can be attached to aluminum. This design is shown in
Figure 5.2. The second option, which we will call the graphite extension, also utilizes
graphite, and is shown in Figure 5.3. This option allows lower electrical heat generation,
but is easier to manufacture than the “fin” as well as making the inner diameter of the
facility constant over its entire length, good for sample insertion and removal. Both
options increase the cross-sectional area, decrease the resistance to electric current, and therefore, decrease the electrical heating outside of the heater region. The larger the cross-sectional area, the lower the electrical heating, but the higher the neutron and gamma heating, since they are proportional to the mass. The size of the graphite region connected to the heater must be determined by balancing these two heating effects. Ultimately, the graphite extension was the option chosen. Both the “fin” and the extension fulfill the purpose of cooling the graphite so it can be attached to aluminum. The extension, however, will be much easier to manufacture. The “fin” would have required a quick change in the diameter of the graphite between the heater and the “fin”. The extension, on the other hand, allows us to use a simple method to connect the heater and the extension – screw threads.

At the end of the heater nearest the tip of the port, the thin walled cylinder of the heater becomes a solid cylinder to decrease heat production at the end. From the end of that solid graphite cylinder, we must provide a way to cool the graphite further, as well as to connect the graphite heater section to the outer pipe. Both objectives are met by using a disc which both the graphite cylinder and the outer pipe attach to, as shown in both Figure 5.2 and Figure 5.3. To make the manufacture and construction of the facility’s tip simpler, the heater, solid cylinder, end disk, and a short section of the outer pipe will be made from one piece of graphite. This end piece can be replaced to change the maximum heating rate by using a different cross-sectional area of the heater section in the replacement.
Figure 5.2
Tip of facility, with “fin” section
Figure 5.3
Tip of facility with graphite extension piece.
5.3. Analysis tools

5.3.1. Analytical approximations

The thermal system at the tip of the sample holder is two dimensional (r,z geometry) with different heat generation in different regions due to electrical heating as well as neutron and gamma heating. This problem is too complex to solve analytically, but using greatly simplified models will allow us to estimate the heat transfer. Neglecting the neutron and gamma heating, we will be able to estimate the maximum electric power needed from the graphite heater. To further simplify the analytical model, electrical heating was only included in the heater section. Neglecting electrical heating outside the heater is a good approximation, as heating is proportional to the electrical resistances shown in Table 5.1.

With the heat generation simplified to a point where it can be dealt with analytically, we have to deal with the two-dimensional heat transfer. For a first approximation, we can analyze the system as:

- a radial heat transfer problem in the region of the heater,
- a fin for the graphite cooler section,
- heat transfer out the hemisphere on the end of the port, assuming temperatures for the hemisphere's base and outer surface, and,
- a convection boundary condition at the outside of the port in the D₂O.

Estimation of the Heat Transfer Coefficient to the D₂O

The D₂O is constantly being circulated to the heat exchanger, so there will be some forced convection on the surface of the port. The volume flow rate of the D₂O is known, as well as the cross-sectional area for flow in the region including the port.[K3,M4] These will give an approximate velocity over the port, which we can use to estimate the Reynold’s number. Using tabulated values for the Prandtl number, we can use an empirical correlation for the Nusselt number as a function of the Reynold’s number and the Prandtl
Finally, we can back out the heat transfer coefficient from the Nusselt number. The calculated heat transfer coefficient for forced convection is $h_{\text{forced}}$, equal to 144 $w / m^2K$. Appendix A contains the calculation leading to this.

Since the velocity of the $D_2O$ over the port is quite low, natural convection can have a potentially large effect. Using the properties of $H_2O$ (but correcting the density for the different molecular weight of $D_2O$) and an empirical correlation for natural convection over a horizontal pipe, we can estimate the heat transfer coefficient for natural convection. This will depend strongly on the wall temperature of the port. If the wall temperature is 50°C, $h_{\text{nat\'1 conv}}$ is estimated to be 579 $w / m^2 K$. If the wall temperature is just 5° higher, 55°C, $h_{\text{nat\'1 conv}}$ equals 741 $w / m^2 K$. The calculations leading to these results are in Appendix A.

*Heat Transfer out the Hemisphere Tip of the Port*

For a simple estimate of the heat transfer through the hemisphere, we can assume that the heat transfer in the hemisphere is purely axial, with all heat transfer occurring parallel to the axis of the port. Appendix C includes the calculations for this analysis of the heat transfer through the hemisphere. If the disk at the base is at 900°C, and we use the heat transfer coefficient for natural convection on the surface of the hemisphere, we calculate that less than 20 watts will go through the hemisphere axially. In reality, the heat transfer will not be purely axial. By neglecting curvature of heat flux lines toward paths of least resistance, we are underestimating the heat transfer. Our estimate should be good within a factor of 2, and we will see that the heat loss through the hemisphere is small relatively to the heat loss from the port wall around the heater.

*Heater and Fin Analysis Method*

The radial heat transfer in the heater section and the analysis of the fin are straightforward problems to solve. Two types of insulation were considered for the region
surrounding the heater. The first is a felt made of ceramic fibers, which can be wrapped around the heater. To solve the heat transfer in this case just requires the dimensions and the conductivity of the insulating felt, which is available from the manufacturer, Zircar®.[Z2] The other type of insulation considered was radiation shields. If convection between the shields is neglected, solving for the heat transfer with radiation shields is reduced to a straightforward problem. To solve for the radiative heat transfer, the emissivity of the materials must be known. For graphite, we can estimate the emissivity to be 0.5, the emissivity for a carbon filament.[L1]

In order to allow iterations of the calculations, programs were written in C to solve the heat transfer equations. This allowed, for example, iteration between the heater power and the base temperature of the fin.

The fin program, FinRad, solved for the heat flux through the fin. FinRad solves the fin equations formulated for a radiation boundary condition, taking as input the base temperature. For the radiation boundary, the emissivity was assumed to be 0.5 and the view factor was 1, appropriate for the geometry of concentric cylinders. The calculations were performed for the cases of radiating to an outer surface at 50°C and 60°C. There was no difference between the calculations for the two different outer temperatures. For this reason, we can safely neglect the temperature difference through the outer pipe, insulation and port wall. Figure 5.4 shows the heat dissipated by the fin as a function of base temperature as well as the temperature of the fin 0.2 m from the base. For base temperatures in the range 800-900°C, we expect to have about 100 watts leave the heater section through the fin.
Figure 5.4
Heat dissipated by the fin, and the temperature 0.2 m from the fin's base.
The program to solve for radial heat transfer with radiation shields, Newton, takes as input the number of radiation shields. The program assumes a heater power, and also assumes an infinite heat transfer coefficient to the D_2O, which is included by setting the temperature at the outside of the port to the D_2O temperature, 45°C. The output of the program is the radial temperature profile of the port. The case of three radiation shields with emissivities equal to 0.5 and 120 watts total power yields a fuel centerline temperature of 1028°C.

The emissivities used for Newton are high—we expect radiation shields to have emissivities on the order 0.1. For this reason, the heat loss from the heater in the radial direction is overestimated. In fact, if we use more radiation shields, we will closer approach the heat transfer for pure conduction through helium. This implies that we can design the facility in such a way that we can neglect heat transfer in the radial direction from the heater. This leads to a axially parabolic temperature distribution in the heater region, shifted slightly, since we expect different amounts of heat to leave by the fin and the base disk.

It should be reiterated that these analytical solutions are very rough estimates. They were used to qualitatively investigate the relative heat transfer through different sections of the facility. One thing we can get from this analysis is that the heat transfer in the radial direction from the heater can be quite small. Radiation shields yield an effective conductivity on the order of the conductivity of ceramic fiber insulations. It is therefore necessary to chose between thermal radiation shields and ceramic fiber insulation based on merits other than insulating value. Radiation shields have two primary benefits, both linked to the small amount of material necessary to make them. Thermal radiation shields will not experience much neutron and gamma heating, and they will not strongly attenuate the neutrons entering the facility. Ceramic fiber insulation has the primary advantage of
being commercially available. Ceramic fiber insulation will therefore be much less expensive. Ceramic fiber insulation has a low density, and the neutron and gamma heating in this insulation is acceptably low. The benefits of thermal radiation shield are small, and the potential cost savings of using ceramic fiber insulation is large. We will therefore consider only ceramic fiber insulation for the remainder of the thermal analysis.

5.3.2. **ALGOR® Computer Code**

Detailed thermal design of the facility was done using a commercial FEM heat transfer code. The thermal analysis module of the ALGOR® finite element analysis code was used. It is capable of handling heat generation in all of the regions of the port tip due to neutron and gamma heating, as well as due to electrical heating. The model built in ALGOR used axisymmetric elements, with an insulated boundary condition along the inner surface of the heater section as well along the inner surface of the graphite extension region. The model included only the region of the port which extends into the D₂O tank. This end point for the model was based on earlier analytical calculations which suggested that the heat transferred along the port or facility beyond that point would be negligible. The boundary condition chosen to end the model was an insulated surface. This will tend to increase the temperature predicted by ALGOR at the end of the model near this surface, but as noted before, little heat transfer was expected beyond that point, and the effect should be small. The solution of the model verified that the thermal gradient is small at the end of the model, so the insulating boundary condition is a conservative but close estimate. The ALGOR model assumed that the space surrounding the heater was filled with Al₂O₃–SiO₂, to a plane 4 cm from the point where the heater and the extension connect.

Heat generation rates used in ALGOR are based on the corrected values calculated by the code MCNP, as discussed in Chapter 4, Neutronics. MCNP calculated the heat generation in ten regions along the axis of the port. Using an exponential curve fit to the
corrected MCNP data, the heating rates can be expressed as a function of position along the axis of the port. Only 16 different heat generation rates can be set in ALGOR. To use the curve fit to calculate heat generation, the file which the decoder sends to the processor must be modified. A short program was written to read the decoder output, determine the location of each of the elements and modify the heat generation in the element according to the curve fit for the MCNP data. The boundary condition applied to the surface which is in contact with the D₂O is convection. The temperature of the D₂O is 45°C and the heat transfer coefficient is the one calculated in Appendix A for natural convection. The problem must be iterated to use the appropriate heat transfer coefficient for the wall temperature.

Two cases were analyzed using ALGOR to determine the heater power required at two extremes of the operating envelope. The first case assumed no neutron and gamma heating, i.e. the case of the reactor at zero power. The second case assumed the reactor was operating at full power, 5 MWth, and the associated neutron and gamma heating was included. Each of these cases was run a number of times to determine the heater power needed to maintain the fuel sample at 1000°C. Another case analyzed determined the temperature profile in the facility when there is only neutron and gamma heating.

Results

When the reactor was at zero power, the heater requirement was approximately 300 watts. Based on the electrical properties of the graphite and the dimensions of the heater, this can be achieved with a power supply of 138 amps, 2.2 volts. When the reactor is at full power, the heater requirement is only 250 watts. Figure 5.5 shows the temperature profile in the model for the case with the reactor at zero power, while Figure 5.6 shows the temperature profile for the case with the reactor at full power. ALGOR also predicted that the maximum temperature reached due to neutron and gamma heating alone was 273°C.
Figure 5.7 shows the temperature profile when there is only neutron and gamma heating, no electrical power.

The calculations based on radiative heat transfer in Newton and finRad suggest that heat transfer in the radial direction from the heater will be negligible compared to the heat leaving through the fin and through the graphite disk on the bottom of the heater. The calculations done with ALGOR confirm this. Figure 5.8 shows the magnitude of the heat flux in different elements. The highest heat fluxes are in the graphite sections, the heater, the extension, the disk at the bottom, and the graphite outer pipe.
Figure 5.6
Temperature profile, electric heating of 250 watts, neutron and gamma heating from reactor at 5 MWth.
Figure 5.7
Temperature profile, no electric heating, neutron and gamma heating from reactor at 5MWth.
Figure 5.8
Magnitude of the heat flux, for the case of 250 watts electric power, neutron and gamma heating from MITR-II at 5MWth.
6. Material choices

The materials used in the facility will be subjected to extreme thermal conditions, as well as large doses neutron and gamma radiation. With the temperature of the sample at about 1000°C and the outer surface of the port at 45°C, there will be large thermal gradients in and around the heater. With the fast neutron flux on the order of $10^{12}$ neutrons/cm$^2$-sec, heating and activation could be substantial. In addition, the large gamma flux at the port will cause heating in the facility. Minimizing the heating due to radiation interactions with the materials of the facility will enhance the safety by reducing the maximum temperature the facility can attain without electrical heating. Certainly, the maximum temperature the facility can attain without electrical heating should be below the temperature at which the structural integrity of the fuel and of the facility in general are in question. When the facility is at the end of its useful life, it will have to be removed and will be radioactive waste. The less activation which has occurred, the easier the facility will be to handle. Also, if the activation is kept to a minimum and if contamination from fission products can be kept small, the facility will not be classified as high level waste, which will simplify its disposal. If, during the life of the facility, it must be removed to be worked on, the less activated it is, the easier and safer it will be to work on the facility.

There are a number of materials which exhibit little activation and heating in neutron and gamma fluxes. To keep the activation as low as possible, we need to choose
materials with very small neutron capture cross-sections. This will also minimize the attenuation of the neutrons before they reach the sample. To minimize heating due to gamma ray interactions, a low atomic-number element is preferable. Materials which have small neutron capture cross-sections and low atomic numbers include graphite, various alloys of aluminum or titanium, and materials composed of aluminum, silicon, carbon, and oxygen. Aluminum-6061, graphite, and an insulation of aluminum oxide and silicon dioxide were chosen. The following sections explain how these choices were made.

6.1. Al-6061

Using a metal for a large portion of the facility will simplify construction. The MITR-II core utilizes aluminum-6061 extensively as a structural material, as well as in the fuel, so there is a well-established database for this material at MIT. Aluminum is fairly easy to machine, and can be purchased in the form of a pipe with a number of diameters available, so the construction of the facility will be eased by the machinability as well as the commercial availability. Unlike titanium, aluminum has a relatively low atomic number, and will experience less gamma heating than most other metals. Aluminum has a low neutron absorption cross-section, so activation and neutron heating are kept to a minimum.

The one serious drawback to using an aluminum alloy in this facility is its relatively low melting point. Al-6061 melts in the range 582-652°C. Since any aluminum in our facility will not be heavily loaded, it is reasonable to limit the maximum aluminum temperature to 300°C. This limitation excludes aluminum from use too near the sample, where the temperature will be well in excess of the operating limit set for the aluminum.
6.2. **Graphite**

Graphite was chosen for the high temperature applications in the fuel test facility. There are a number of materials which can be used for electrical resistance heating to relatively high temperatures, including graphite, molybdenum, tungsten, silicon carbide, molybdenum disilicide, nickel-chromium alloys and iron-chromium-aluminum alloys.[M3] The iron based alloys can be eliminated from consideration since they become activated in a neutron field. The nickel based alloys have melting temperatures in the range 1350-1400°C. For a heater operating at 1000°C, this is rather low. Molybdenum and tungsten both have high tensile strength, but their electrical resistivity is 10 to 100 times lower than the ceramics included in the initial list. To produce the heat needed in molybdenum or tungsten, the current necessary would be very high, or the cross-sectional area would have to be extremely small. For this reason, we eliminate molybdenum and tungsten from consideration for the heater.

All of the materials we are left with—graphite, molybdenum disilicide, and silicon carbide—can have operating temperatures in inert atmospheres between 1600°C and 2200°C. Of these three, molybdenum disilicide has the highest tensile strength but the lowest resistivity. Silicon carbide has a higher resistivity and lower tensile strength than molybdenum disilicide. Graphite has the highest resistivity of the three (100 times higher than the resistivity of silicon carbide) and the lowest tensile strength (10 times lower than the tensile strength of molybdenum disilicide). To get the necessary heat production in the heater, the higher resistivity of graphite makes the difference between a difficult to manufacture heater and an impossible to manufacture heater.
Summarizing the beneficial characteristics of using graphite as an electrical heating element:
- graphite experiences little activation in a neutron flux,
- graphite can be operated to 2200°C in an inert atmosphere,
- graphite has higher resistivity than the other choices, which leads to larger heater cross-section and lower required current.

6.3. Insulation

There are a great variety of insulations commercially available. Considering only those which are stable above 1000°C reduces the options substantially. Additionally, only those insulations composed of elements with low neutron absorption cross-sections can be considered, further narrowing the possibilities. Our application benefits from choosing an insulation with a high specific insulating value, that is, a large thermal resistance per unit mass. Neutron and gamma heating is proportional to the mass of a material, so the less mass the insulation has, the less heating there will be in it, the better it will be able to perform its function — to insulate the heater.

An insulation composed of 95% Al₂O₃, 5% SiO₂, marketed by Zircar® and called alumina paper, is the insulation recommended.[Z2] This insulation meets all the requirements described above. In addition, it is bought in flexible sheets which can easily be molded to fit our facility. Another advantage of this insulation is that it is available with no organic binder. This is especially good since it will be used in a closed facility, where any noxious gases produced when organic binders are set would be difficult to vent.
Chapter 7

7. Control and Measurement

7.1. Temperature

The temperature of the sample must be carefully measured and controlled. The temperature of the sample is one of the variables that is very important for proper analysis of the experimental results. A circuit that will regulate the power of the heater based on the measured temperature of the sample should be designed.

To measure the temperature of the sample, optical pyrometry is recommended. There are a number of advantages to optical pyrometry. The first is that there is no need to insert a probe along the length of the port – the sensor can be placed at the end of the port where access is easier. In the temperature range of interest for the sample, 600°C - 1000°C, optical pyrometry is quite accurate. The TRIGA King Furnace used optical pyrometry, as have other experiments involving radioactive samples at very high temperatures. The equipment is commercially available, so there is no concern about getting appropriate equipment.

The drawbacks of optical pyrometry result from the geometry of our facility. The sample size to port length ratio is quite small, so this application would require relatively high quality optics and mirrors, with accurate sighting. This will be challenging, since there must be a way to get the samples in and out without ruining the optics’ accuracy. The TRIGA King Furnace solved this by putting a mirror on the outside of the facility, apparently with a “window” through the facility. The mirror assembly and the “window”
are removed as a unit from the facility, allowing entry to the facility without disturbing the relative position of the mirror and window. A system similar to this can be designed for the Fuel Test Facility.

One commercial manufacturer of optical pyrometry equipment, Mikron®, markets a system utilizing optical fibers to transmit the temperature information. Applying this technology to the Fuel Test Facility will help avoid neutron streaming problems. Mikron has developed optical fibers which are resistant to radiation damage, unlike ordinary optical fibers which are damaged quickly by radiation. The applicability of product should be investigated.

7.2. Atmosphere

The atmosphere in the facility will be similar to that in an HTGR. This will serve two purposes. The first is to minimize chemical reactions between the graphite at very high temperatures and impurities in the gas. The second purpose of maintaining an atmosphere similar to that in an HTGR is that it will eliminate any uncertainty about the potential effect of the gas on the performance of the fuel. Bates et al. use the impurity levels shown in Table 7.1 in their experiments on metals in reactor helium.[B2] It will be most important to control the H₂ / H₂O ratio, since that controls the oxidizing potential of the gas. This is typically done by flowing the gas over a copper oxide bed, causing the hydrogen to react and form water. The gas stream is then passed through a filter to remove the water vapor. To measure the impurities in the gas, a gas chromatograph can be used. A small amount of the gas can be sent to a gas chromatograph for measurement. The atmosphere in the facility will be kept slightly above atmospheric pressure to eliminate in-leakage of air. Since the pressure will be above atmospheric, a make-up system will be necessary to maintain the pressure if there is any leakage.
Table 7.1. Impurity levels in reactor helium used by Bates et al.[Ref Bates]

<table>
<thead>
<tr>
<th>Impurity</th>
<th>Partial Pressure (μatm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>H₂</td>
<td>50 - 100</td>
</tr>
<tr>
<td>CO</td>
<td>25 - 50</td>
</tr>
<tr>
<td>CH₄</td>
<td>3 - 8</td>
</tr>
<tr>
<td>H₂O</td>
<td>0.5 - 3</td>
</tr>
<tr>
<td>O₂</td>
<td>Below Detection Limits</td>
</tr>
<tr>
<td>N₂</td>
<td>≤ 11</td>
</tr>
</tbody>
</table>

Since this will be a fuel experiment, it will be necessary to provide a system to monitor the activity in the gas. This will tell us if any particles have failed during irradiation. Monitoring for activity in the helium can be done by diverting a small flow of helium to a liquid nitrogen cooled charcoal trap. The gamma activity of the trap can then be measured constantly. [S2]
8. Sample Handling

There are a number of considerations when designing the sample and the equipment to handle it. The sample must be large enough to hold as much fuel as we want irradiated in one experiment. The sample must be small enough to fit in the heater. There can be some iteration between the sample size and the heater size, but limits on the amperage of the heater and limits on the minimum thickness which can be machined will set a maximum inner diameter for the heater. Additionally, the sample will be made of graphite, as the heater is. The sample and the heater must be electrically insulated, or the cross-section for the electric current will be much larger in the region where the heater and sample are in contact. This will decrease the heating around the sample, and put a large current through the sample, both of which we want to avoid.

During irradiation, the sample may be at temperatures of 1000°C. This is too hot to put in contact with many materials, so there will be a cooling down time after irradiations before samples are removed. That will also give some time for the short-lived fission products to decay, decreasing the radiation shielding requirements.

8.1. Activity of sample after irradiation

Clearly, one of the major concerns about handling specimens is their radioactivity after irradiation. In order to protect experimenters, the samples must be shielded when they are removed from the facility. Other than the fuel in the sample, the primary material is graphite. There will also be some insulation. These are materials which do not activate
strongly. As a result, the radioactivity of the sample will be due almost entirely to the
decay of fission products. The activity of the fission products will depend mainly on three
variables— the fission rate, the irradiation time and the time since the irradiation ended.

Glasstone and Sesonke present a semi-empirical correlation to estimate the beta activity
of spent fuel as a function of operating power, irradiation time and cooling time.[G1] The
correlation is based on the assumption that one beta particle is emitted for every fission
product decay. Their correlation, accurate to within a factor of two for cooling times from
10 seconds to 100 days, is

\[
\text{Activity} = 14 \times P_\circ \left( \tau^{0.2} - (\tau + t)^{0.2} \right) \text{[Ci]}
\]

where

- \( P_\circ \) is the power of the particle in watts,
- \( \tau \) is the cooling time in seconds, and
- \( t \) is the irradiation time in seconds.

Assuming that the irradiation is continuous, i.e. no shutdowns of the reactor on weekends,
we can calculate the irradiation time for a given burn-up based on the burn-up estimates
discussed in Chapter 4. Additionally, we can estimate the power per particle from the
neutronics calculations. Once the burn-up is specified, the correlation yields the activity in
Curies as a function of cooling time. Figure 8.1 shows the activity of one particle as a
function of cooling time for 10% FIMA and 20% FIMA. Since our samples will have on
the order of 20 particles, these values of activity would have to be multiplied by a factor of
20 to be accurate for the whole sample.
8.2. **Inserting and removing samples**

Insertion and removal of the samples pose unique problems. The facility is in a high radiation field, and all sample handling at the facility will have to be done from behind radiation shielding. Additionally, any handling equipment must ensure that we know the position of the sample at all times, and that we not allow the sample to fall from our grip. The major problem, which magnifies all other problems, is that the sample is irradiated at the end of a 1 cm diameter pipe about 3 meters long.

Insertion and removal require fundamentally different performance characteristics, so two unique tools have been designed for these applications. The inner diameter of the facility is constant for the length of the facility, which simplifies the tasks. Insertion of the sample is relatively simple: the sample must be pushed to the end of the facility, and must stay there once the insertion tool is removed from the facility. The first goal, pushing the sample to the end of the facility, can be accomplished with a long rod whose end is flat, so that it can push on the fuel. By measuring the length of the facility, a mark can be put on the insertion rod that shows how far the rod must be inserted to insure that the sample is at the tip of the facility. The second insertion goal is to be certain that the sample remains at the tip of the facility. This goal can be achieved rather simply by taking advantage of gravity. There is sufficient room radially to tilt the entire facility a few degrees so that the heater end is slightly lower than the end of the facility away from the reactor. Then, once the sample has been pushed in, there will be no force to push the sample uphill, and its location will remain constant.

Removal of the sample is not as simple. Since the sample must be pulled out, it is necessary to maintain a hold on the sample. If the sample should become separated from the removal tool, it would be more difficult to grab it again. We also must be sure that
there will be a way to let go of the sample so it can be placed in a shielded container. Due
to the difficulty of working with a very long pole to try to remove the sample, we have
designed a tool which does not require accurate alignment. The tool, shown in Figure 8.2
with the sample, can be pushed into the hole in the sample's end. Once inserted, four
“fingers” will spring back out to their normal, slightly flared position, locking on the
sample. The sample can then be pulled from the facility. Finally, we must address the
issue of how the removal tool deposits the sample in a shielded container. As designed, the
end of the removal tool, including the “fingers” and a short shaft, ending in a piece of
square cross-section, will be left attached to the sample. This will provide a handle of sorts
for working with the sample after irradiation, as well as simplifying the task of leaving the
sample shielded. The removal tool will fit through a square hole in the end of a shielded
container for the sample. The container will be placed up against the end of the facility, and
the removal tool will be used to pull the sample into the container. The square hole in the
container will allow the tool handle to slide freely, but will just fit the square cross-section
of the tool’s end. The end of the tool will be attached to the pole by screw threads, and the
square hole will act as a wrench to hold the square shaft while the tool is removed from the
pole. The details of the shielding container have to be worked out, but this idea should
prove simple and reliable.

Both insertion and removal require that the tool be inserted the full length of the
facility, or about 3 meters. Since there is not much space outside the port, it would be
beneficial to find a way to shorten the length of the tool outside the port when the sample is
at the outer end of the facility. A potential solution is to use a rod that is jointed instead of
rigid. If the rod is jointed at 1 meter intervals, then only 1 meter is need behind the
shielding which eases the tight fit in the area outside the port. The joints should lock in the
straight position so the joints do not fold between the shielding and the port.
Sample

Removal Tool

Section B-B

All dimensions in mm.

Figure 8.2
Sample and removal tool.
Chapter 9

9. Future Work

The scope of this thesis has been limited by time and funding. While major design issues were addressed, only a few of these issues were dealt with in great detail. With this in mind, this section is intended to provide a summary of the areas which still need substantial attention. There are two categories of work which needs to be done. The first is additional work on the design of the fuel test facility, and the second is investigation of related facilities.

9.1. Continued Work on the Fuel Test Facility

Much of this thesis deals with the thermal design of the test facility. No experimental data, however, was available on neutron and gamma heating in the port. A computer model of the reactor was used to estimate the heating rates, but the results of that analysis were not conclusive. It is possible that after the completion of this document, the computer model's predicted heating rates will be shown to agree with experimental data available from other facilities in the reactor. If this is the case, heating rates predicted by the computer model can be used with confidence. If this is not the case, the author recommends that a simple experiment be conducted to measure the neutron and gamma heating in the D₂O region.

The experiment suggested is a relatively simple one which will provide simple, but accurate information. The port this facility has been designed for is 6RH2. There is an identical port mirrored about the thermal column, called 6RH1. 6RH1 has been fitted with
a pneumatic rabbit system. The proposed experiment would use this rabbit system. A piece of graphite, shaped according to the requirements of the rabbit system, with a thermocouple attached to it, could be put into the rabbit system manually. The pneumatic system could be disconnected during the experiment to minimize convective heat transfer. Then, if the graphite is insulated, the rate at which the graphite heats up can be used to calculate the heating rate in the port. This is the most accurate way to determine the heating in 6RH2 without opening that port. Once conclusive data on heating is available, the thermal design of the facility can be verified using existing models with appropriately modified heating rates. In addition to providing information for the design of the fuel test facility, this experiment will provide another set of data to check the results of MCNP by.

Once a complete, accurate thermal model of the facility is developed, the sample size should be modified so that the fueled region is in the region of the heater with the highest temperatures. Current models show the hottest part of the heater to be about 3 cm from the base of the heater. To put the fueled region of the sample in this region would just require an extension of the unfueled graphite region by about 2.5 cm. This is the easiest way to place the fuel region in the hottest part of the heater.

The design of the gas seal system for the facility, as well as the electrical connections have not been adequately addressed yet. Requirements for these components have been suggested in the sections on sample handling and on temperature measurement. The design of the TRIGA King Furnace provides one method for accomplishing these goals, and it should be kept in mind when considering sealing the system and completing the electrical circuit.

The shield plug currently on the port will need to modified to allow for access to the facility. This may include room for the temperature measuring system within the shielding.
Also, the design of the shield plug will have to be coordinated with the design of a shielded container for removal of the sample.

9.2. Further work on space external to the port

During design of the fuel irradiation facility, the interaction of the facility to the area external to the port has not been addressed. During the irradiation phase of any experiment, the only interaction of the facility with the space outside the port will be a need for space for the power supply for the heater, instrumentation for temperature measurement, and a system to maintain an inert atmosphere in the facility. When samples are changed, however, there will be more activity outside the facility which will have to be coordinated with the equipment which is external to the port as well as equipment which is not associated with this facility but which is placed in the area around the port. Shielding for experimenters will be necessary, and there will have to be provisions made to open the port from behind shielding, as well as to manipulate samples from behind shielding. The author has looked at the region outside the port and believes that this is possible with no major changes to the layout of the region.

9.3. Related Facilities

There are a number of areas which should be investigated to complete the overall system design, of which this experimental facility is only one. Other facilities are necessary to build a complete experimental program. These include a facility to coat particle fuels on site and post-irradiation facilities to examine the fuel.

Since the kernels for the sample particles would most likely be uniform, and since the kernels are somewhat more complicated to manufacture, it might be advantageous to purchase the kernels pre-formed. Having a facility to put the coatings on kernels, however, would allow experimenters to make just a few particles with well-known coating parameters. Experimenters would be able to make new samples with coatings tailored to
investigate specific phenomena quickly, upon learning the results from previous experiments. There would be no delay while an outside contractor produced the samples. In addition, it is doubtful that there is any commercial manufacturer of coated particle fuels who would be interested in repeatedly making very small batches of fuel. With no reactors in the United States currently using coated particle fuel, the ability to manufacture it here at MIT would be a significant advantage.

On-site post-irradiation examination facilities are vital to maintaining momentum in sample analysis so new experiments can be focussed appropriately. The "Irradiated-Microsphere Gamma Annalyzer", termed the IMGA system, was developed and used at Oak Ridge National Laboratory, and provides an example of what is possible. This facility analyzes the gamma spectrum of individual particles and uses the spectrum information to determine the retention rates of specific radionuclides. The IMGA system was designed to handle "large populations of irradiated particles". Analysis batches can contain 5000 particles. This is much more than would have to be analyzed at MIT, but the system provides an excellent example of the type of post-irradiation examination facility which could be used at MIT to gain the most information possible from the irradiations.
Chapter 10

10. Conclusions

This work has shown that it is possible to design a facility to perform coated particle irradiations with sample temperatures to 1000°C without exceeding the thermal limits of the MIT reactor. Such a facility would be a valuable addition to the Nuclear Reactor Laboratory’s facilities, and would position MIT as a leader in coated particle fuel research. While there are still many details which need to be addressed before this design can be constructed, a solid basis from which to proceed is now available.

A number of issues are still of concern. The first is the relatively low fast fluence that the fuel samples will receive. There is no way to increase the fast fluence in the 6RH2 port, so if this threatens to overly limit the usefulness of this facility, a more appropriate location may be needed. Heating rates in the port were estimated using a numerical simulation because no experimental data was available. An experiment to confirm the simulation’s predictions should be performed. Licensing questions were not addressed by this work. The licensability of the fuel test facility should be investigated before any further work is done on its design.
Appendix A

A. Burn-up Estimates.

We have an estimate of the flux levels in the center of the heater which we can use to estimate how long it will take to reach specified burn-ups for the fuel samples. We are given that:

\[ \Phi_{\text{thermal}} = 5.67 \times 10^{13} \text{ neutrons/cm}^2 \text{ sec} \]
\[ \sigma_{\text{fission, thermal}} = 549 \text{ barns} = 549 \times 10^{-24} \text{ cm}^2 \]
\[ \text{kernel diameter} = D = 500 \times 10^{-6} \text{ m} \]
\[ \rho = 10.81 \text{ g/cm}^3 \text{ for UO}_2 \]

We must assume that the energy limits on this thermal flux (E < 1.9 \times 10^{-6} \text{ MeV}) are appropriate for use with the tabulated thermal fission cross-section for uranium-235 (for E = 0.025 \text{ eV}). This is only an approximation. For more detailed calculations, the MCNP model should be used to predict the fission rate.

First, we have to calculate the mass of the kernel, and use the mass to determine the initial metal loading.

\[ \text{Volume} = \frac{4}{3} \pi \left( \frac{D}{2} \right)^2 \rho = 7.075 \times 10^{-4} \text{ kg/kernal} \]

If we assume 10% enrichment, we can determine the molecular weight of UO\(_2\) by:

\[ 0.1 \left( 235 \frac{\text{g}}{\text{mol}} \right) + 0.9 \left( 238 \frac{\text{g}}{\text{mol}} \right) + 2 \left( 16 \frac{\text{g}}{\text{mol}} \right) = 269.7 - \frac{\text{g}}{\text{mol}} 10\% \text{ enriched UO}_2 \]

Now, to determine the initial metal loading, we need to take the mass of the kernal, divide by the molecular weight, and multiply by Avogadro's number, as:

\[ \text{IMA} = 7.075 \times 10^{-4} \frac{\text{g}}{\text{kernal}} \times \frac{\text{mol}}{269.7 \text{ g}} \times \frac{6.02 \times 10^{23} \text{ molecules}}{\text{mol}} \times \frac{1 \text{ U atom}}{\text{molecule}} \]

\[ = 1.58 \times 10^{18} \text{ initial metal atoms per kernal!} \]
To determine the fission rate per particle for the given flux, multiply the number of U-235 atoms in the kernel by the fission cross section and the thermal flux.

\[
\text{\ Fissions \!} = \frac{0.10 \times 1.58 \times 10^{18} \text{ \ U \ atoms \!}}{\text{sec} \cdot \text{kernel}} \times \frac{549 \times 10^{-24} \text{ cm}^2}{\text{Kernel}} \times \frac{5.67 \times 10^{13} \text{ neutrons}}{\text{cm}^2 \cdot \text{sec}}
\]

\[
= 4.92 \times 10^9 \frac{\text{\ fissions}}{\text{sec} \cdot \text{kernel}}
\]

For 1\% FIMA, we need 1.58\times10^{16} fissions, which we can use with the fission rate to determine the time to reach 1\% burnup:

\[
1.58 \times 10^{16} \frac{\text{fissions}}{\text{kernel}} \times \frac{\text{sec} \cdot \text{kernel}}{4.92 \times 10^9 \frac{\text{fissions}}{\text{sec}}} \times \frac{\text{hr}}{3600 \text{ sec}} \times \frac{\text{day}}{24 \text{ hrs}}
\]

\[
= 37.2 \text{ days per 1\% FIMA}
\]

10\% FIMA requires 372 days, just over a year of full power days. To reach 20\% FIMA will take about 743 days.
Appendix B

B. Estimation of Heat Transfer Coefficients

B.1. Heat Transfer Coefficient for Forced Convection

This section shows the calculation to estimate the heat transfer coefficient from the port wall to the D\textsubscript{2}O flowing over it. Given:

\begin{itemize}
  \item D\textsubscript{2}O volume flow rate \(= \dot{V} = 150 \text{ gal / min} = 9.48 \times 10^{-3} \text{ m}^3 / \text{sec} \)
  \item D\textsubscript{2}O tank outer radius \(= r_{\text{out}} = 0.615 \text{ m} \)
  \item D\textsubscript{2}O tank inner radius, at the port height \(= r_{\text{in}} = 0.2795 \text{ m} \)
  \item Average D\textsubscript{2}O temperature \(= T_{\text{avg}} = 45^\circ \text{C} \)
  \item Port Diameter \(= D_p = 0.0826 \text{ m} \)
\end{itemize}

Throughout this appendix, the subscript \(D\) will be used to denote properties of D\textsubscript{2}O, while the subscript \(H\) will be used for H\textsubscript{2}O.

Flow area \(= \pi (r_{\text{out}}^2 - r_{\text{in}}^2) = 0.973 \text{ m}^2 \)

Velocity of the D\textsubscript{2}O \(= v_D = \frac{\dot{V}}{A_D} = \frac{9.48 \times 10^{-3}}{0.943} = 1.01 \times 10^{-2} \text{ m/s} \)

Now, to calculate the Reynold’s number based on diameter, \(Re_D\), we will use:

\[
Re_D = \frac{\rho_D v_D D}{\mu_D}
\]

where

\(\rho_D = \) density of D\textsubscript{2}O at 45\(^\circ\)C,
\(\mu_D = \) viscosity of the D\textsubscript{2}O at 45\(^\circ\)C.

The density of H\textsubscript{2}O, at 45\(^\circ\)C, is 990 kg / m\(^3\). We know the ratio of the molecular weight of H\textsubscript{2}O to the molecular weight of D\textsubscript{2}O, and if we assume that the ratio of the densities is the same, we can calculate the density of the D\textsubscript{2}.
\[ \rho_D = \rho_H \left( \frac{m(D_2O)}{m(H_2O)} \right) = 990 \frac{kg}{m^3} \left( \frac{20}{18} \right) = 1100 \frac{kg}{m^3} \]

Assume that the kinematic viscosity, \( \nu \), of D\(_2\)O is the same as that of H\(_2\)O, which we can get from standard tables as 0.566x10\(^{-6}\) m\(^2\)/s for water at 45\(^{\circ}\)C. Then, the viscosity is the product of the kinematic viscosity and the density.

\[ \mu_D = \nu \rho_D = (0.566x10^{-6})(1100) = 6.226x10^{-4} \frac{kg}{m\cdot s} \]

We now have all the terms necessary to calculate the Reynold's number:

\[ Re_D = \frac{\rho_D v_D D}{\mu_D} = \frac{(1100)(1.01x10^{-2})(0.0826)}{6.226x10^{-4}} = 1474 \]

If we assume that the Prandtl number of D\(_2\)O is the same as that of H\(_2\)O, we can use a table to get \( Pr = 3.67 \)

The Reynold's number and the Prandtl number can be used in an empirical correlation to determine and average Nusselt number based on diameter, \( \overline{Nu_d} \), according to the following expression [L1]

\[ \overline{Nu_d} = 0.3 + 0.62 \frac{Re_d^{1/2} Pr^{1/3}}{[1+(0.4 / Pr)^{2/3}]^{1/4}} = 35.3 \]

By definition,

\[ \overline{Nu_d} = \frac{h D}{k} \]

where \( k \) is the thermal conductivity of D\(_2\)O. We will use the thermal conductivity of H\(_2\)O, which is 0.6367 w/m K. Then,

\[ \overline{h} = \frac{\overline{Nu_d} k}{D_p} = \frac{(35.3)(0.6367)}{0.0826} = \boxed{272 \frac{w}{m^2 K}} \]

### B.2. Natural Convection Heat Transfer Coefficient

To estimate the heat transfer coefficient for natural convection, we can use an empirical correlation. Lienhard, [L1], presents such a correlation for the case of flow over a horizontal cylinder, which corresponds to the geometry of the facility. For these
calculations, the properties of H₂O will be used for the properties of D₂O. The error this will introduce due to the difference in molecular masses is on the order of 10% and is an acceptable error. The heat transfer will be strongly dependant on the wall temperature, since the wall temperature is the force which changes the fluid properties locally, driving the flow and the heat transfer. Calculations will be presented here for the case of the wall temperature equal to 50°C.

Given:

\[ T_{\text{wall}} = 50^\circ\text{C} \quad \text{wall temperature} \]

\[ T_{\text{bulk}} = 45^\circ\text{C} \quad \text{bulk temperature of the D}_2\text{O} \]

\[ g = 9.8 \text{ m/s}^2 \quad \text{acceleration due to gravity} \]

\[ D_p = 0.0826 \text{ m} \quad \text{outer diameter of the port} \]

\[ \rho_{\text{bulk}} = 990 \text{ kg/m}^3 \quad \text{density of the D}_2\text{O at the bulk temperature} \]

\[ \rho_{\text{wall}} = 987.8 \text{ kg/m}^3 \quad \text{density of the D}_2\text{O at the wall temperature} \]

\[ \Delta T = T_{\text{wall}} - T_{\text{bulk}} = 5^\circ\text{C} \]

\[ T_{\text{avg}} = \frac{T_{\text{wall}} + T_{\text{bulk}}}{2} = 47.5^\circ\text{C} \]

\[ \frac{1 - \rho_{\text{bulk}}}{T_{\text{wall}} - T_{\text{bulk}}} = 4.393 \times 10^{-4} \text{ K}^{-1} \]

Coefficient of thermal expansion \( \beta \equiv \frac{1 - \rho_{\text{bulk}}}{T_{\text{wall}} - T_{\text{bulk}}} \)

From tables, we can get values for \( \alpha \), the thermal diffusivity, \( \nu \), the kinematic viscosity, and \( k \), the thermal conductivity, all evaluated at 45°C, the average temperature:

\[ \alpha = 1.541 \times 10^{-7} \text{ m}^2 / \text{s} \]

\[ \nu = 0.566 \times 10^{-6} \text{ m}^2 / \text{s} \]

\[ k = 0.6367 \text{ W/m K} \]

The empirical correlation for the Nusselt number which we can use to determine the heat transfer coefficient, is based on the Raleigh number, Ra, and the Prandtl number, Pr. The Prandtl number for H₂O is tabulated, and for the average temperature, it is 3.67. The Raleigh number, based on the outer diameter of the port, is given by:
\[
\text{Rad} = \frac{8 \beta \Delta T D_p^3}{\alpha \nu} = 1.388 \times 10^8
\]

The correlation for the average Nusselt number, based on the port diameter is:

\[
\overline{\text{Nu}_d} = \left\{ 0.6 + 0.387 \left[ \frac{\text{Rad}}{\left( 1 + [0.559 / \text{Pr}]^{9/16} \right)^6} \right]^{1/6} \right\}^2 = 75.03
\]

Now, we can use this value of the average Nusselt number to calculate the average value of the heat transfer coefficient, \( \overline{h} \) :

\[
\overline{h} = \frac{k \overline{\text{Nu}_d}}{D_p} = \begin{bmatrix} 579 \text{ w/m}^2\text{K} \end{bmatrix}
\]

As mentioned above, this value depends strongly on the wall temperature. For the case where the wall temperature is 55ºC, \( \overline{h} = 741 \text{ w/m}^2\text{K} \).
Appendix C

C. Heat Loss through the Hemisphere at the end of the Port.

This section will detail an estimate of the heat loss through the hemisphere on the end of the port using the method of thermal resistances. The total resistance includes conductive and convective resistances. Since the resistances are in series, the total resistance is the sum of the individual resistances. Assume the heat flow is limited to the axial direction. This will lead to an underestimate of the heat transfer.

The basic equation for using resistances to calculate heat transfer is

\[ Q = \frac{\Delta T}{\sum R_i} \]

where:
- \( Q \) is the heat transferred,
- \( \Delta T \) is the temperature difference across the system, and
- \( R_i \) is the resistance for each type of heat transfer, \( i \).

Our geometry is shown in Figure C.1. All space between the end of the facility and the port wall will be filled with insulation. We will assume that the insulation is in direct contact with the D\(_2\)O, which is equivalent to assuming infinite thermal conductivity for the aluminum of the port. This is reasonable since the conductivity of the aluminum is 1000 times greater than the conductivity of the insulation.

We know that the radius of the hemisphere, \( R_0 \), is 3.81 cm (1.5 inches).
Figure C.1. Schematic of hemisphere for heat transfer.

The resistance to convection for differential cylinder of radius \( r \), thickness \( dr \), is given by:

\[
    dR_{\text{conv}} = \frac{1}{h \, dA(r)}
\]

where:
- \( h \) is the heat transfer coefficient to the D\(_2\)O, and
- \( dA(r) \) is the differential area for heat transfer to the D\(_2\)O as a function of radius.

The resistance to conduction is similarly given by:

\[
    dR_{\text{cond}} = \frac{L(r)}{k \, dA(r)}
\]

where:
- \( L(r) \) is the height of the differential cylinder, and
- \( k \) is the thermal conductivity of the insulation.

\[
    dR_{\text{conv}} = \frac{1}{h \, (2 \, \pi \, r) \, R_o \ d\theta} = \frac{1}{h \, 2 \, \pi \, R_o^2 \, \cos \theta \ d\theta}
\]
\[
\frac{dR_{\text{cond}}}{(2\pi r)dr} = \frac{1}{kR_o \cos \theta d\theta}
\]

\[
dR_{\text{total}} = dR_{\text{conv}} + dR_{\text{cond}} = \frac{1}{2\pi R_o} \left[ \frac{k + hR_o}{khR_o \cos \theta d\theta} \right]
\]

For a disk with a radius of 2.54 cm, we can calculate \( \alpha \) to be equal to 48.2°. Integrate the differential total resistance from \( \theta \) equals 0 to \( \alpha \).

\[
\int_0^\alpha \frac{d\theta}{R_{\text{total}}(\theta)} = \frac{2\pi R_o^2 k h}{(k + hR_o)} \int_0^{48.2^\circ} \cos \theta d\theta = \frac{2\pi R_o^2 k h}{(k + hR_o)} (0.7455)
\]

\[
Q = \frac{\Delta T}{R_{\text{total}}} = \Delta T \left( \frac{1}{R_{\text{total}}} \right) = \Delta T \left[ \frac{2\pi R_o^2 k h}{(k + hR_o)} (0.7455) \right]
\]

Inserting the following values, calculate the total heat transfer to be 15 watts.

- \( k = 0.1 \text{ w/m °C} \)
- \( h = 580 \text{ w/m}^2 \text{ °C} \)
- \( R_o = 2.54 \times 10^{-2} \text{ m} \)
- \( T_{\text{disk}} = 900^\circ \text{C} \)
- \( T_{\text{heavy water}} = 45^\circ \text{C} \)
- \( \Delta T = 855^\circ \text{C} \)
References


H1. Hanson, D. Personal communication to C.M. Martin, Sept 1990.


M4. MITR blueprints, numbers: R3G-V-4, R3S-1-5, R3G-7-5.


