A ONE-GROUP PARAMETRIC SENSITIVITY ANALYSIS FOR THE

GRAPHITE ISOTOPE RATIO METHOD AND OTHER RELATED

TECHNIQUES USING ORIGEN 2.2

A Thesis

by

KRISTIN ELAINE CHESSON

August 2007

Major Subject: Nuclear Engineering

A ONE-GROUP PARAMETRIC SENSITIVITY ANALYSIS FOR THE

GRAPHITE ISOTOPE RATIO METHOD AND OTHER RELATED

TECHNIQUES USING ORIGEN 2.2

A Thesis

by

KRISTIN ELAINE CHESSON

Submitted to the Office of Graduate Studies of
Texas A&M University
in partial fulfillment of the requirements for the degree of

MASTER OF SCIENCE

Approved by:

Chair of Committee,   William S. Charlton
Committee Members,   Charles F. Hermann
                       Pavel Tsvetkov
Head of Department,   John Poston

August 2007

Major Subject: Nuclear Engineering

ABSTRACT

A One-Group Parametric Sensitivity Analysis for the Graphite

Isotope Ratio Method and Other Related Techniques

Using ORIGEN 2.2. (August 2007)

Kristin Elaine Chesson, B.S., Texas A&M University

Chair of Advisory Committee: Dr. William S. Charlton

Several methods have been developed previously for estimating cumulative energy production and plutonium production from graphite-moderated reactors. The Graphite Isotope Ratio Method (GIRM) is one well-known technique. This method is based on the measurement of trace isotopes in the reactor's graphite matrix to determine the change in their isotopic ratios due to burnup. These measurements are then coupled with reactor calculations to determine the total plutonium and energy production of the reactor. To facilitate sensitivity analysis of these methods, a one-group cross section and fission product yield library for the fuel and graphite activation products has been developed for MAGNOX-style reactors. This library is intended for use in the ORIGEN computer code, which calculates the buildup, decay, and processing of radioactive materials. The library was developed using a fuel cell model in Monteburns. This model consisted of a single fuel rod including natural uranium metal fuel, magnesium cladding, carbon dioxide coolant, and Grade A United Kingdom (UK) graphite. Using this library a complete sensitivity analysis can be performed for GIRM and other techniques. The sensitivity analysis conducted in this study assessed various input parameters including ^{235}U and ^{238}U cross section values, aluminum alloy concentration in the fuel, and initial concentrations of trace elements in the graphite moderator. The results of the analysis yield insight into the GIRM method and the isotopic ratios the method uses as well as the level of uncertainty that may be found in the system results.

NOMENCLATURE

1DB	One-dimensional neutron diffusion code
Al	Aluminum
B	Boron
BOL	Beginning-of-life
CANDU	Canadian Deuterium Uranium Reactor
Cl	Chlorine
DANT	Diffusion Accelerated Neutron Transport code
DOE	Department of Energy
ENDF	Evaluated Nuclear Data File
g	Grams
GIRM	Graphite Isotope Ratio Method
JEFF	Joint Evaluated Fission and Fusion Library
JENDL	Japanese Evaluated Nuclear Data Library
kg	Kilograms
Li	Lithium
MAGNOX	Magnesium oxide
MCNP	Monte Carlo N-Particle Transport Code
MONTEBURNS	Automated, Multi-Step Monte Carlo Burnup Code System
MT	Metric Ton (1000 kg)
MTU	Metric Ton of Uranium
MWe	Megawatts Electric
MWt	Megawatts Thermal
NJOY	Code System for Producing Pointwise and Multigroup Neutron and Photon Cross Sections from ENDF/B Data.
NNDC	National Nuclear Data Center
ORIGEN	Oak Ridge Isotope Generation and Depletion Code
PNNL	Pacific Northwest National Laboratory

ppm	Parts per million
Pu	Plutonium
S	Sulfur
Ti	Titanium
U	Uranium
UK	United Kingdom
V	Vanadium
w/o	Percent by weight
WIMS	Winfrith Improved Multip-group Scheme integral transport code
XS	Cross section

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

CHAPTER I

INTRODUCTION

Verifying the production of fissile material in nuclear facilities is a key element in the efforts of nuclear nonproliferation. The production of fissile material can result from several processes. Some of these processes include the enrichment of uranium for fuel to power nuclear reactors, the production of plutonium within the fuel of a reactor during operation, and the buildup of ^{233}U from ^{232}Th via the irradiation of thorium. During these processes some evidence is left behind that could lead an investigator to predict the most likely events from the past that would have led to the evidence observed in the present day. This study of evidence and its relationship to past material production is often known as "nuclear archaeology" and is rooted in the verification of nuclear weapons activities.[1]

Several methods have been developed previously for estimating cumulative energy production and plutonium production from graphite-moderated reactors.[2-4] The Graphite Isotope Ratio Method (GIRM) is one well-known technique. The GIRM technique was originally developed in the 1990s as a joint venture between the United States and Russia as a method to evaluate the large uncertainty associated with the Russian plutonium production during the history of its weapons program. The program, funded by the U.S. Department of Energy (DOE), was initiated at Pacific Northwest National Laboratory (PNNL) but involved collaboration with other laboratories, universities, and Russian institutes during the development and evaluation of the GIRM method.[5] This work spanned nearly a decade in length and focused on the validity of the use of GIRM to estimate plutonium production in graphite reactors—most specifically the graphite-moderated reactors found in Russia. However, the method is applicable to any graphite-moderated reactor. Reactors of this type have been operational in several countries throughout the world.[6] These include:

This thesis follows the style of the *Journal of Nuclear Materials Management.*

- Calder Hall and other reactors in Great Britain

- G1 and G2 Marcoule reactors and others in France

- Production reactors in Russia

- Hanford production reactor in US

- The "5-MWe" production reactor in DPRK

- Tokai 1 reactor in Japan

- Vandellos 1 reactor in Spain

- Latina reactor in Italy

Nuclear and Reactor Physics Principles

Neutrons can interact with matter in several different ways, and these interactions can be divided into the two main categories of scattering and absorption. Scattering collisions do not change the physical composition—the number of protons or neutrons—of the nucleus off which the incident neutron scatters. Absorption reactions, on the other hand, can emit a variety of particles or induce fission. These events do change the physical structure of the atom with which it interacts. For this reason, it is the absorption reactions which are of interest for the application of GIRM. Several different emissions can result following the absorption of a neutron by a nucleus. Charged particle emissions occur when the nucleus rids itself of a proton (n,p), alpha particle (n,a), or deuteron (n,d) following the absorption of a neutron. In other cases where excess neutrons are released from the nucleus, only the emission of more than one neutron can be recognized as a different reaction from scattering such as (n,2n) and (n,3n) reactions. It is also possible for the neutron to be captured by the nucleus. The new nucleus, which now possesses one more neutron than before, is left in an excited state and emits a gamma ray to return to a stable ground state. This is known as a radiative capture reaction (n,?). A fission reaction may also occur, (n,f), which results in several emissions including fission fragments, neutrons, and gamma rays.[7]

The basic nuclear physics principle of GIRM is that atoms undergo predictable changes during neutron irradiation. The neutron fluence in a reactor is defined as the time integral of the neutron scalar flux. The evaluation of the changes caused by the exposure of a neutron fluence to the natural uranium fuel, graphite moderator, or other structural materials provides some measure of the fluence that caused those changes. The energy produced in the core of the reactor is a direct measure of the number of fission reactions that have occurred. For each fission reaction approximately 2.4 neutrons are produced. Of these neutrons, one must be absorbed in U-235 to cause a subsequent fission to maintain the critical chain reaction. Some fraction of the remaining neutrons is absorbed in U-238 to produce plutonium, and some other fraction is absorbed in other materials such as the moderator and structural materials. The absorption of neutrons by the trace elements found in the graphite moderator causes shifts in the isotopic composition of those trace elements. Modern reactor physics codes such as WIMS, MCNP, DANT, and 1DB can accurately predict the behavior of the neutrons and the fractions of neutrons absorbed in the various structural components of the reactor.[8-11]

For use as a moderating material in nuclear reactors, a material must possess the characteristic that it is highly scattering with a low cross section for neutron absorption. Graphite possesses this characteristic; however, graphite ore could have significant traces of other highly absorbing elements, such as boron. Thus, for use in nuclear reactors, only high-purity graphite is used in which the other mineral impurities are kept below some defined tolerance. The determination of these maximum concentrations depends largely on the values of the neutron absorption cross sections for the element's isotopes. Some common impurities that can be found in reactor-grade graphite are boron, chlorine, titanium, lithium, vanadium, zinc, and uranium. Figure 1 and Figure 2 display the expected changes to natural boron and natural uranium when irradiated.

Figure 1. Boron burnup in a reactor
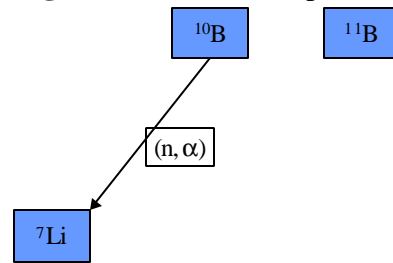
Figure 2. Plutonium buildup in a natural uranium fueled reactor
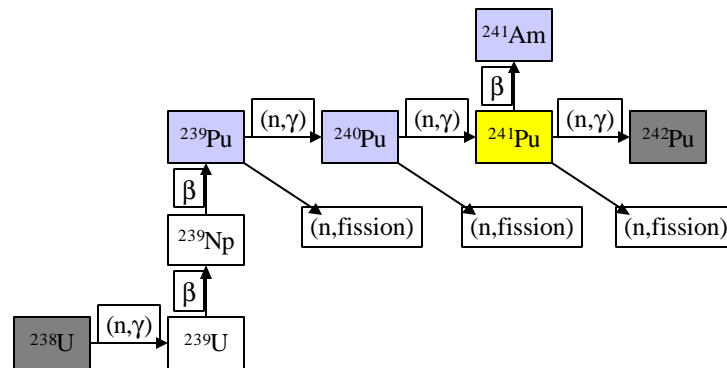
Since the initial concentrations of the graphite impurities prior to irradiation are often unknown, it is difficult to correlate fluence to the measurement of an isotope's concentration after irradiation. However, by measuring a ratio of two isotopes, it is possible to determine the fluence without knowing the exact concentration of either isotope at beginning of life. Taking the transmutation of boron shown in Figure 1 as an example, the concentration of B-10 and B-11 isotopes can be represented by simple balance equations. Neglecting the absorption of neutrons in B-11 the resulting balance equations are given by:

$$N^{10}(t) = N_0^{10}(t) \cdot \exp\left[-\boldsymbol{s}_a \int_0^t \boldsymbol{f}_{graph}(t')dt'\right] \tag{1}$$

$$N^{11}(t) = N_0^{11}(t) \tag{2}$$

where $N^{10}(t)$ and $N^{11}(t)$ represent the B-10 and B-11 concentrations, respectively, in the graphite as a function of time; N_0^{10} and N_0^{11} represent the initial concentrations prior to irradiation of the B-10 and B-11 isotopes, respectively; the microscopic absorption cross section for B-10 is s_a; and $\phi_{graph}(t')$ is the neutron scalar flux at time t' in the graphite.

If equation 1 is divided by equation 2, we acquire

$$\frac{N^{10}(t)}{N^{11}(t)} = \frac{N_0^{10}(t)}{N_0^{11}(t)} \cdot \exp\left[-\boldsymbol{s}_a \int_0^t \boldsymbol{f}_{graph}(t')dt'\right] \tag{3}$$

The neutron fluence, F_{graph}, is defined as the time integrated neutron scalar flux and can thus be represented by the following:

$$\Phi_{graph}(t) = \int_0^t \boldsymbol{f}_{graph}(t')dt' \tag{4}$$

Substituting equation (4) into equation (3) yields:

$$N^{10/11}(t) = N_0^{10/11}(t) \cdot \exp\left[-\boldsymbol{s}_a \Phi_{graph}(t)\right] \tag{5}$$

We can solve this for the neutron fluence:

$$\Phi_{graph}(t) = -\frac{1}{\boldsymbol{s}_a} \ln\left[\frac{N^{10/11}(t)}{N_0^{10/11}(t)}\right] \tag{6}$$

where the ratio of the B-10 to B-11 atoms in the graphite sample at any time, t, is given by $N^{10/11}(t)$ and the variable $N_0^{10/11}(t)$ represents the ratio of the atoms prior to irradiation. The ratio of B-10/B-11 atoms prior to irradiation is the ratio of B-10/B-11 found in nature and is assumed to be known. Thus, by measuring the B-10/B-11 atom ratio in the graphite at time t, the fluence of neutrons can be directly determined.

If the fluence in the graphite can then be related to the fluence in the fuel, the cumulative energy produced in the reactor can be determined according to:

$$E(t) = E_R \int_{V_f} N_f s_f \Phi_{fuel}(t) dV \tag{7}$$

where E_R is the recoverable energy per fission (approximately 200 MeV), N_f is the concentration of fissile atoms in the fuel which is presumably all U-235, $F_{fuel}(t)$ is the fluence in the fuel at time t, the variable s_f is the microscopic fission cross section, and V_f is the volume of all of the fuel. If we assume that the fluence in the fuel can be represented by some volume averaged fluence [$\overline{\Phi}_{fuel}(t)$] at any time t, then we would have

$$E(t) = E_R \overline{\Phi}_{fuel}(t) \Sigma_f V_{fuel} \tag{8}$$

where S_f is the volume- and time-averaged macroscopic fission cross section and V_{fuel} is the volume of the fuel.

The fluence in the graphite can be related to the fluence in the fuel with a lattice physics calculation for a reactor pin cell:

$$R(t) = \frac{\overline{\Phi}_{fuel}(t)}{\overline{\Phi}_{graph}(t)} \tag{9}$$

If the ratio of these fluxes were constant with time then we could say

$$R = \frac{\overline{\Phi}_{fuel}(t)}{\overline{\Phi}_{graph}(t)} \tag{10}$$

By substitution of equations (7) and (10) into equation (8), we would then have

$$E(t) = -E_R \frac{R}{s_a} \ln\left[\frac{N^{10/11}(t)}{N_0^{10/11}(t)}\right] \Sigma_f V_{fuel} \tag{11}$$

Thus, by measuring the B-10/B-11 ratio and performing a simple pin cell calculation to acquire R and S_f, we can acquire the cumulative energy produced by the system. Empirically we find that the assumption that R is time-dependent is poor in many cases. Thus, instead of using equation (11), we can calculate the expected B-10/B-11 ratio versus cumulative energy production directly from the pin cell model. This cumulative

energy production can be directly related to the fuel burnup in that pin cell and to the quantity of plutonium produced in the fuel in that pin cell.

Description of the Graphite Isotope Ratio Method

The GIRM method is useful as a verification tool for the plutonium production of graphite-moderated reactors when little or no access is available to the spent fuel. Ideally, the best assessment of the plutonium produced in a reactor is the direct measurement of the spent fuel by mass spectroscopy. However, such methods could be costly, time-consuming, or impossible due to inaccessibility to the spent fuel. GIRM provides an inference of this quantity with no required samples or measurements of the fuel, but rather through measurements of the isotopic ratios of trace elements in the graphite. Properly chosen ratios are useful for estimating the plutonium produced in the reactor regardless of the length of time following the reactor shutdown.

Several elements were determined to be suitable for use as indicators for the measurement of reactor burnup when using the GIRM method based on factors that include the measurable quantity of the impurity in the graphite, the activation products of the elements, and the cross sections of the different isotopes. These elements, as listed in the GIRM Primer, are shown in Table 1.[4,12] Ratios of chlorine isotopes have also been identified as useful indicator ratios in the GIRM analysis.[12] The cross sections of the isotopes must lead to a meaningful change in the ratios after irradiation and produce other stable activation products. The concentrations of the trace elements also must be large enough that sufficient quantities can be measured from a graphite sample using mass spectroscopy methods. The behavior of the isotopic ratios with changing neutron fluence determines the range for which the ratio is best suited as a fluence indicator. This fluence is directly related to the burnup of the fuel as well as the plutonium and energy production in the reactor as shown above.

Table 1. GIRM indicator ratios

Measured Isotopic Ratios	Applicable Fluence Range
B-10/B-11	Low
Li-6/Li-7	Low to Intermediate
Ti-48/Ti-49	Intermediate to High
U-235/U-238, U-236/U-238	Low to High
Pu-240/Pu-239, Pu-241/Pu-239, Pu-242/Pu-239	Low to High

Sample Measurements

Impurities in the graphite come from the environment when graphite ore is mined from the earth. Failed fuel rods can also be a source of impurities. Even with concentrations at parts per million levels, the graphite impurities are measurable with mass spectroscopy. The first step in GIRM is to acquire a large enough representative set of samples for analysis by mass spectroscopy such that the fluence map of the core can be estimated. Several samples are drilled from a range of locations in the reactor core using commercially available machinery. Locations of the samples can be at nearly any axial location in the core, from any fuel channel, and at any depth into the graphite. The optimization of the sampling methodology, which includes the location and quantity of samples taken, minimizes the error in the global plutonium estimate.

The sample preparation, which is not described here but can be found in detail in reference 13, must be completed such that the risk of environmental contamination is minimal.[13] Fuel rod failures can also contaminate the graphite. Since the neutron flux is different in the fuel and graphite, measurement of isotopic ratios for elements which originated in the fuel will not be representative of the fluence found in the graphite. However, these events will be documented in the reactor's operational history, and such samples can be avoided. Various isotopic ratios can then be measured using one of three different mass spectroscopy methods: Secondary Ionization Mass Spectroscopy,

Thermal Ionization Mass Spectroscopy, or Inductively Coupled Plasma Mass Spectroscopy.[4] The spectroscopic method to be used depends on the ratios which are being measured, and require different procedures in the sample preparation.

Local Energy and Plutonium Production Estimates

       The next step of GIRM involves fuel cell calculations using a reactor physics code. Some available reactor physics codes are the Winfrith Improved Multi-group Scheme (WIMS) integral transport code, the Monte Carlo N-Particle transport code (MCNP), Diffusion Accelerated Neutron Transport code (DANT), and 1DB, a one-dimensional neutron diffusion code.[8-11] A fuel cell model consists of one fuel rod and the surrounding materials in the fuel channel which include the cladding, carbon dioxide coolant, and graphite moderator. Figure 3 displays a horizontal cross-sectional view of a general fuel cell for graphite-moderated reactors. The fuel cell calculation provides the uranium fuel rod burnup in terms of megawatt-days per kilogram of fuel (MWd/kg), which is the time integrated reactor power per unit mass of fuel. This computation also generates the relationship between the fuel isotopes and the trace isotopes in the graphite that are immediately adjacent to the fuel rod.

Figure 3. Cross sectional view of fuel cell
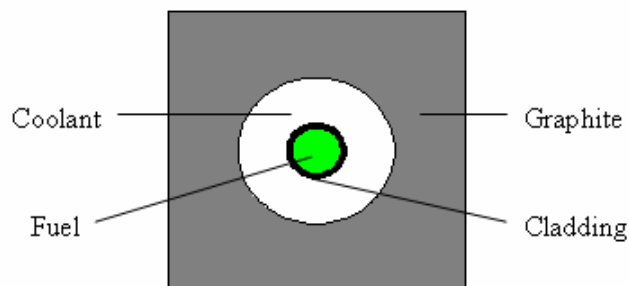
The fuel cell calculations are local plutonium estimates for each graphite sample. The model determines the expected isotopic ratios in graphite as a function of fuel burnup and the plutonium produced in the fuel as a function of burnup. This information establishes a direct and quantitative correlation between the isotopic ratios and plutonium content. Using this correlation, the measured ratio from a graphite sample, for instance B-10/B-11, can estimate the energy and plutonium produced at the specific location associated with the sample.

Figure 4 and Figure 5 display examples of the information obtained from a fuel cell calculation. Figure 4 shows the change of the B-10/B-11 and the U-235/U-238 ratios as a function of fuel burnup. The change of U-235 and Pu-239 masses with respect to the reactor burnup is in Figure 5. Thus, if the measured value of B-10/B-11 from a graphite sample is 0.04, the approximate fuel burnup for that specific sample location is 1000 MWD/MTU according to the results in Figure 4. The corresponding plutonium production for that specific location obtained from Figure 5 is approximately 10 grams of Pu-239. Repeating this process for each available sample produces a set of local plutonium estimates throughout the reactor. This set is reasonably small compared to the number of fuel channels within the reactor since only selected fuel channels are sampled. The number and location of the samples are optimized for each reactor.

Figure 4. Dependence of selected trace isotopes to reactor burnup
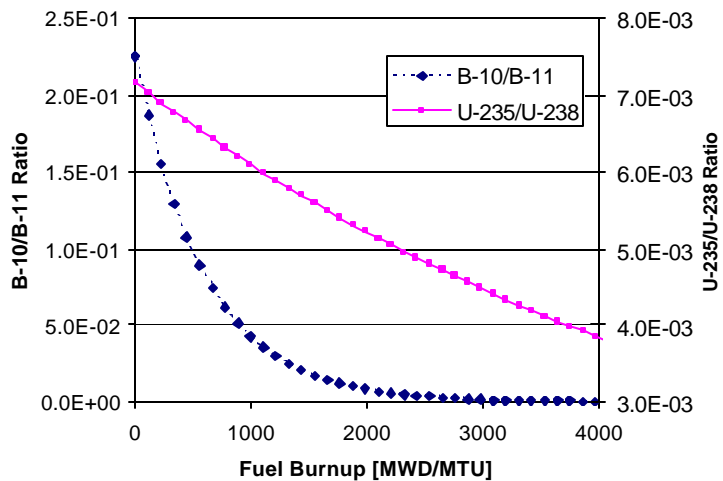
Figure 5. Fuel rod inventory during burnup (initial fuel inventory of 12.4 kg)
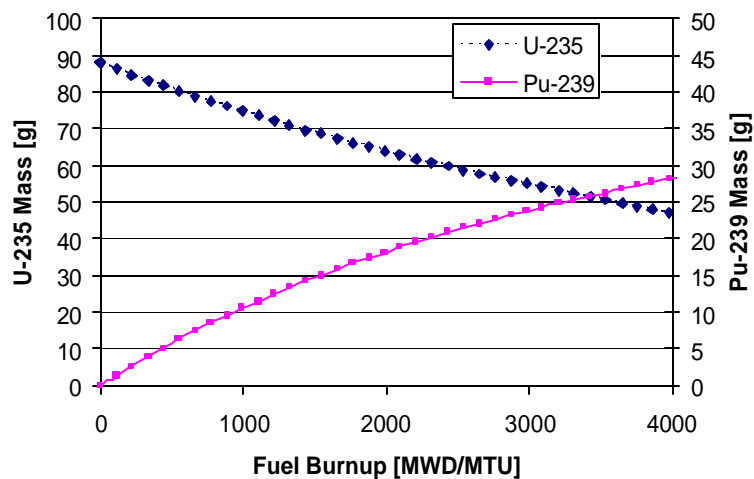
Global Energy and Plutonium Production Estimates

The next step in GIRM includes using the set of local fuel burnup values with a regression analysis technique to fit a series of basis functions to the measured data. The

result is a three-dimensional, full-core fluence profile defined for every point in the reactor which best fits the set of local estimates. The fluence field is assumed to be a linear combination of eigenfunctions. When detailed operational data is unknown, these functions are found to be the eigenfunction solutions to the homogeneous diffusion equation for the core. These equations satisfy the problem boundary conditions and are linearly independent.[14] An appropriate number of basis eigenfunctions must be determined to fit the data for each particular reactor. A complex, less smooth fluence field requires more basis functions for an adequate fit. Once an adequate weighted regression of these functions is developed for the fluence model, the average production of plutonium in the reactor is determined from the integrated fluence over the reactor volume.

When knowledge of the operational history and design of the reactor core is available, a modification to the above regression analysis can be used. The operational information can include power levels, length of operation at each power level, periods of shut down, and control rod positions. A three-dimensional full-core calculation with a lattice-physics code such as WIMS is constructed using this information. The result is a steady-state fluence profile for the reactor core. This profile is then used as the first basis function in the regression analysis. Other subsequent basis functions account for any incomplete knowledge that may exist about the reactor such as the accuracy of information regarding the control rod positions. A more in-depth and detailed description of the regression analysis for the global estimate can be found in reference 14.

CHAPTER II

DEVELOPMENT OF ORIGEN 2.2 DATA LIBRARY

To facilitate a parametric sensitivity analysis of the GIRM technique it was desirable to use a code system which is well-known and has a relatively short computational time. The ORIGEN code satisfies both of these traits. However, the data libraries which are included with the ORIGEN code package are not applicable for a graphite-moderated reactor. Thus, it was necessary to develop a one-group cross section and fission product yield library for MAGNOX-style reactors. This library was developed using Monteburns and was verified for use at average burnups for graphite reactors.

Code Systems Used

ORIGEN, the Oak Ridge Isotope Generation and Depletion code, is a reactor physics buildup and depletion code developed by Oak Ridge National Laboratory.[15] This point-depletion and decay code can be used for calculating the isotopic compositions of fuels and targets in various reactor systems using one-group cross section data. ORIGEN includes standard libraries for many reactor systems including pressurized water reactors, boiling water reactors, CANDU (Canadian-deuterium uranium) reactors, and liquid-metal fast breeder reactors. Among its many features is the ability to calculate isotope concentrations, radioactivity, fission rates, and neutron absorption rates.

ORIGEN 2.2 requires three databases to be input by the user: a radioactive decay library, photon production library, and cross section library.[15] The radioactive decay database supplies the list of nuclides to be included in all calculations with information for these nuclides such as half-lives, branching ratios, recoverable energy per decay for parent nuclides, and isotopic compositions of naturally occurring nuclides. The photon libraries provide the number of photons produced per decay using an established 18-group energy structure. Cross section libraries supply information regarding effective one-group nuclide cross sections as well as fission product yields. These one-group cross

sections are calculated external to the ORIGEN 2.2 code and are reactor and fuel specific with fixed spectra.[16]

The Monte-Carlo N-Particle transport code (MCNP) is written and maintained by Los Alamos National Laboratory.[9] MCNP is a continuous energy, time-dependent, coupled neutron-photon-electron transport code which uses the Monte Carlo method for particle transport. This method tracks each source particle for a given number of source particles from birth to absorption. MCNP is capable of modeling almost any geometry and uses pointwise cross section data. It has the ability of calculating surface currents, flux tallies, energy deposition in a cell, pulse height tallies, and other tallies. The code can also perform criticality calculations. [9]

Monteburns is a linkage code used to link MCNP with the depletion code ORIGEN.[17] The version used for these calculations utilized MCNP5 and ORIGEN 2.2. The code consists of a Perl script which executes both of these codes and links the information between them. To run Monteburns, an MCNP input deck and Monteburns input file are required with the option of an additional material feed file. MCNP primarily provides one-group microscopic cross sections and flux values to the ORIGEN code which then uses these values for burnup calculations. The ORIGEN results are then used to create a new MCNP input deck. A predictor step is calculated with ORIGEN to a point halfway through the burn step. MCNP then uses the information to create one-group cross sections at the burn step's midpoint.[17] This calculation assumes that the isotopics and neutron flux spectrum halfway through the burn step are reasonable for the entire burn step. This assumption is valid and helps to increase the accuracy of the results under the restriction that burn steps are not too long (generally less than 2500 MWD/MTU).

During the execution of Monteburns, several input and output files are created at each burn step. Of specific interest are the ORIGEN cross section libraries which are created as TAPE9_$.I# files where "$" refers to the Monteburns material number and "#" refers to the burn step. At each burn step, the ORIGEN files are modified to contain the new cross-section and flux values obtained by MCNP.[17] For the development of a

one-group cross section library for a MAGNOX-style reactor, this file will be taken and analyzed at a burn step of average burnup for MAGNOX-style reactors.

Depending on the complication of the geometry, Monteburns simulations can take weeks or even months to complete the calculations. The number of times in which MCNP is accessed and the number of particles required for good statistics in each MCNP calculation results in a lengthy computational time. The computational time required to run ORIGEN, however, is very little—on the order of seconds to minutes—making ORIGEN a much more attractive tool to utilize in sensitivity analyses where running many different cases is required.

NJOY is a nuclear data processing system which processes evaluated nuclear data into pointwise or multi-group nuclear cross sections that can be used with computer codes such as MCNP.[18] NJOY also has the ability to treat several nuclear effects such as Doppler broadening of resonances, self shielding, thermal scattering, and many other physical events. The code is capable of evaluating neutrons, photons, and charged particles and produces libraries for use with a variety of particle transport and reactor analysis codes.

Description of the Calder Hall Reactor

The Calder Hall Unit 1 reactor at Seascale was the first large gas-cooled power reactor in the world, owned and operated by British Nuclear Fuels PLC beginning in 1956. This reactor was an outgrowth of the one built at Windscale and was originally used for the production of plutonium. The Calder Hall plant was shutdown in 2003. The Calder Hall reactors are MAGNOX-style reactors. MAGNOX reactors are gas-cooled, graphite-moderated, and use a natural uranium fuel. The fuel cladding is a magnesium alloy, which is the basis for the name MAGNOX.[6,19,20]

Because of the lack of large fuel enrichment facilities at the time of construction, natural uranium was used as the fuel for the Calder Hall reactors. For similar reasons, conventional materials and techniques were employed when building the plant. Conserving neutrons in a natural uranium fueled reactor is crucial for maintaining the

critical chain reaction; thus, the materials used in these cores must have extremely low neutron absorption cross sections. This is the primary reason for the use of graphite as a moderator and low-absorption gases (like carbon dioxide) as a coolant. This is also one of the main reasons for the usage of a uranium metal fuel since the oxygen in oxide fuels increases the parasitic absorption in the fuel.

Of the many materials suitable for cladding purposes, only magnesium satisfied the two requirements of conventionality and low neutron absorption. Magnesium is unfortunately not a high-temperature material with a melting point of about 1200°F and a maximum working temperature of about 850°F. Thus, in the Calder Hall reactor, the maximum cladding temperatures were limited to about 750°F.[20] When the maximum cladding temperature is set in the design, the maximum operating fuel temperature is determined by the fuel-element size and specific power (in W/g) at the point of interest. Natural uranium fuel requires the use of a wide lattice pitch to improve the resonance escape probability and results in the use of relatively large diameter fuel rods. A large diameter fuel element corresponds to a large temperature drop in the fuel; however, the maximum fuel centerline temperature must be limited for safety reasons. Due to this factor, a relatively low specific power and consequently a low core power density is necessitated in this type of reactor. This tends to limit the gas temperatures significantly. These limitations have also led to the use of helical fins on the cladding to increase heat transfer rates. The following design information can be found in references 6, 19, and 20.

Calder Hall Fuel Elements

The Calder Hall Unit 1 reactor operates on MAGNOX fuel elements. These elements are constructed of cast, cylindrical natural uranium metal rods, which are treated and machined to a diameter of 2.92 cm and a length of 101.6 cm. The rods are clad in a 1.5 mm MAGNOX A 12 cladding with helical fins. MAGNOX A 12 is an alloy of magnesium. The fins have a thickness of 0.8 mm, height of 13 mm, and pitch of 3.16 mm. The fuel rods have MAGNOX end caps that seal the fuel rod. To one end of the

fuel rod is attached a cone-shaped structure which is designed to allow for grasping of the fuel rod by a fuel handling tool. On the other side of the fuel rod is a cup designed to seat on top of the cone-shaped end of an adjacent fuel rod. The rods are stacked end-to-end in the fuel channels of the Calder Hall core. A schematic diagram of these MAGNOX fuel elements is shown in Figure 6 below.

Figure 6. Calder Hall Unit 1 fuel element (not to scale, all dimensions in inches)
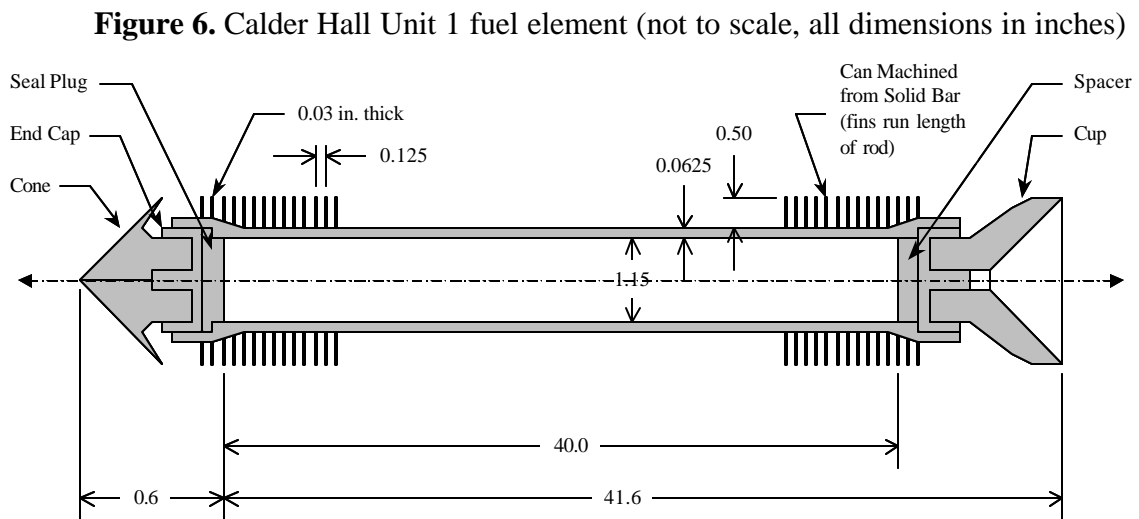
Calder Hall Moderator and Reflector

The Calder Hall Unit 1 reactor is moderated using 650 tons of grade A graphite. Each graphite block has dimensions of $20.32 \times 20.32 \times 81.28$ cm. These moderator bricks are located with keyed tiles. Centered in each moderator brick is a machined fuel element channel with a diameter of 10.0 cm. A schematic of each moderator brick is shown in Figure 7.

The reactor is reflected using 490 tons of grade B graphite composed of 14,000 bricks. These bricks are of the same size as those in the moderator; however, there are no locating tiles between the reflector bricks. The reflector bricks completely surround the core such that the top, bottom, and sides of the reactor are reflected.

Figure 7. Moderator block for Calder Hall Unit 1 (not to scale)
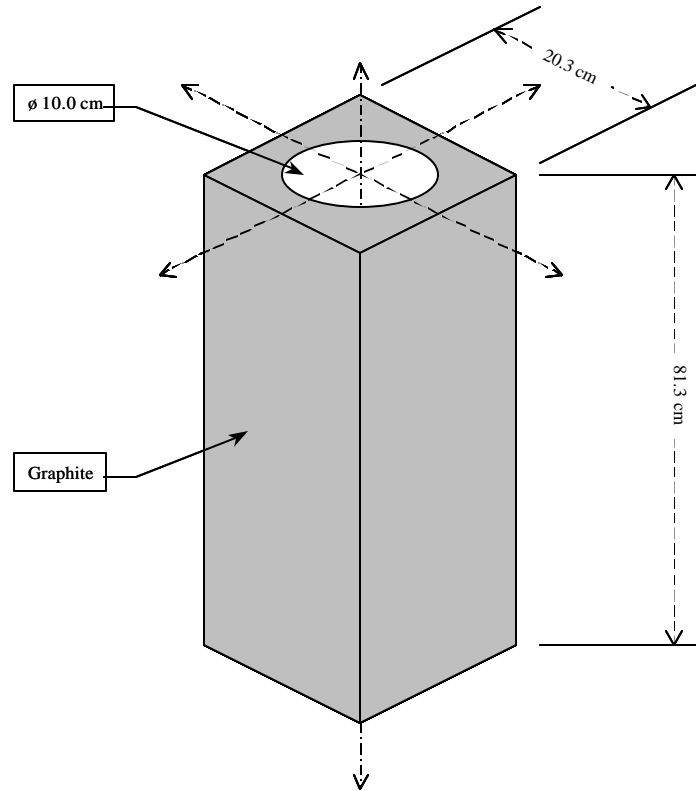
Control Rods

The Calder Hall Unit 1 core is controlled using 48 boron steel control rods in stainless steel tubes. The control rods have a diameter of 4.45 cm and a total length of 7.00 m. Control rod channels with a diameter of 6.00 cm are machined into the moderator blocks as shown in Figure 8. These control rods can be operated in gang, and any four of these control rods can be used as regulating rods. The control rods also serve as the shim safety rods for the reactor. The core also contains fixed absorbers for power shape flattening purposes.

Figure 8. Calder Hall Unit 1 control rod channel arrangement
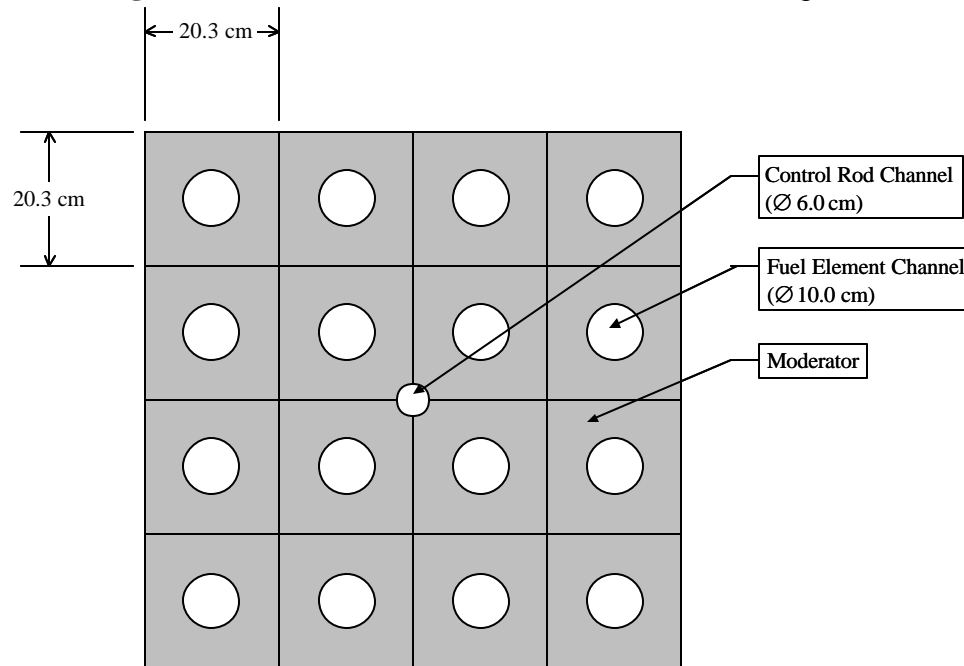
Core Arrangement

The Calder Hall Unit 1 reactor is roughly cylindrical in shape with a diameter of 9.45 m and height of 6.40 m. The core is composed of 650 tons of moderator and 490 tons of reflector. These moderator and reflector bricks are stacked into the cylindrical shape of the core. There are 1696 fuel element channels in the core with six fuel elements stacked end-to-end in each channel. Thus, there are a total of 10,176 fuel elements in the core. The total core loading at rated power is 127 MT of natural uranium. The rods are positioned in a square lattice with a 20.3 cm pitch. A 5-cm thick pressure vessel composed of A1-killed high manganese steel surrounds the entire core. Table 2 contains various parameters for this core. Figure 9 contains a horizontal cross sectional view of the Calder Hall Unit 1 reactor.

The Calder Hall Unit 1 core operates at a net power of 50 MWe or 201 MWt. The average specific power in the fuel is 1.38 W/gU. The average power density in the core is 0.45 kW/liter. The graphite moderator has an average temperature at power of

237°C. The maximum clad temperature at power is 500°C, and the maximum fuel temperature is 600°C.

Table 2. Calder Hall Unit 1 core parameters

Active Core Height	9.45 m
Active Core Diameter	6.40 m
Fuel Enrichment	natural uranium
Fuel Inventory	127 MTU
Average Fuel Specific Power	1.38 W/gU
Average Core Power Density	0.45 kW/l
Average Fuel Element Discharge Burnup	4000 MWd/MTU
Fuel Rod Pitch	20.3 cm
Number of Fuel Rods	10176
Refueling Frequency	18 tHM/year

Figure 9. Calder Hall Unit 1 core horizontal cross section

Gas Inlet
and Outlet

Gas Inlet
and Outlet

Biological Shield
(7 feet thick concrete)

Thermal Shield
(6 inches thick mild steel)

Pressure Vessel
(37 feet, 4 inches O.D.)

Side Reflector

Control Rod Positions

Gas Inlet
and Outlet

Gas Inlet
and Outlet
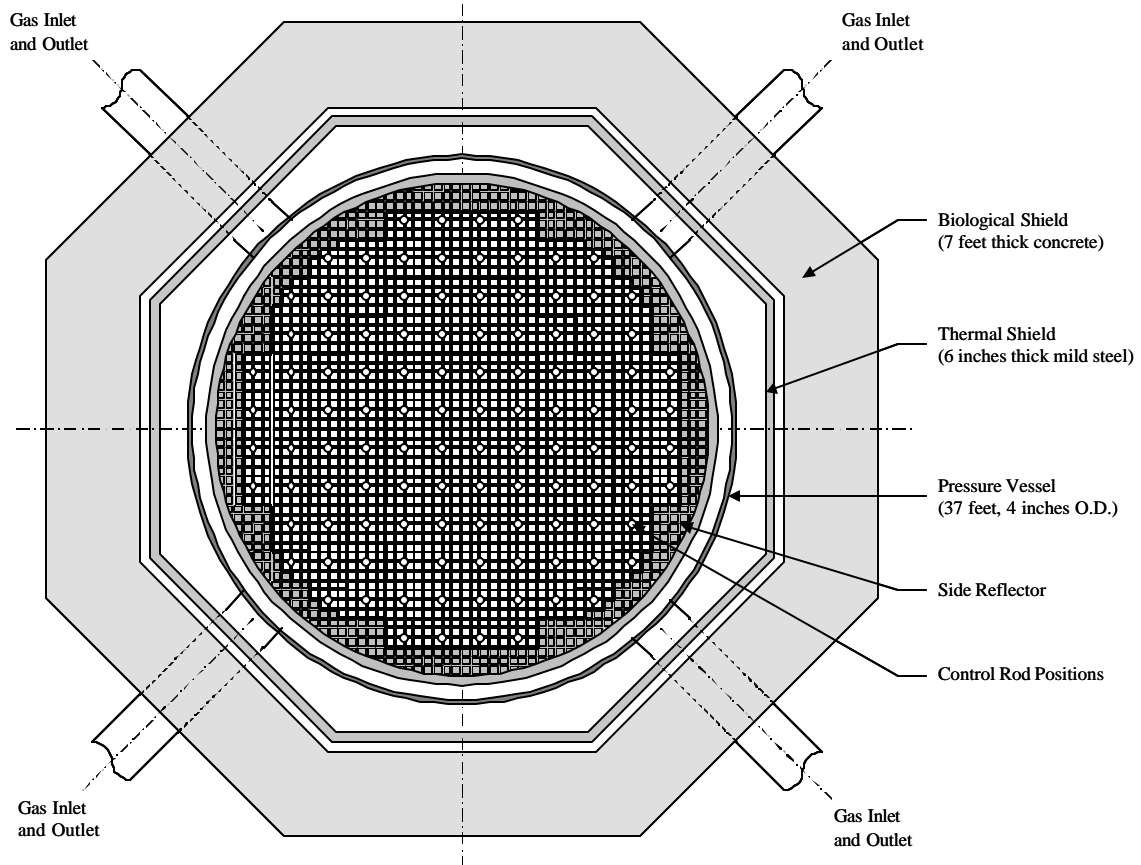
Monteburns Model and Simulation

The fuel cell model used in Monteburns consisted of a single fuel pin with reflective boundaries. The details of the fuel pin are listed in Table 3. Some simplifying assumptions were made to the actual fuel pin design as described above. The helical cladding fins were omitted from the model. These fins increase heat transfer from the fuel pin to the coolant; however, the material is nearly transparent to neutrons. Since the focus of this project is the neutronic analysis and transmutation of elements found in the reactor and surrounding materials, it is reasonable to omit these fins from the pin cell model. For similar reasons, the conical end caps for each rod were modeled as cylindrical end caps of approximately the same volume. Following production of the fuel

elements, a very small gap is present between the uranium fuel pellet and the MAGNOX cladding. During operation, the fuel is heated and thermal expansion occurs closing this gap. It was assumed for the Monteburns fuel pin model that the fuel has thermally expanded and closed the fuel-to-clad gap.

Table 3. Specifications for a Calder Hall fuel pin

Fuel:	Nat. U metal	Fuel length:	101.6 cm
Cladding:	Magnesium alloy	Fuel Pellet Diameter:	2.921 cm
Coolant:	CO_2	Clad Thickness:	0.15875 cm
Moderator:	Grade A UK graphite	Pin-to-Pin Pitch:	20.3 cm
Average Power Density:	1.38 W/g	Fuel Rod Outer Diameter:	3.2385 cm
Fuel Rod Channel Diameter:	10.6 cm	Top & Bottom Clad Plug Length:	2.1167 cm
Fuel Density:	18.17 g/cc	Clad Density:	1.74 g/cc
Coolant Density:	0.013 g/cc	Moderator Density:	1.7 g/cc

The core of the Calder Hall reactor is divided into three zones: (1) an inner zone consisting of 256 fuel channels, (2) an intermediate zone of 576 fuel channels, and (3) an outer zone of 864 fuel channels. These zones have varying fuel channel diameters drilled into the graphite blocks. Inner, intermediate, and outer channel diameters are 10.6 cm, 10.0 cm, and 9.2 cm, respectively.[20] The fuel rod channel for the Monteburns pin cell model was chosen from the inner section of the assembly with a 10.6 cm diameter fuel channel in the graphite block. Figure 10 displays a horizontal cross sectional view of the fuel pin geometry. Figure 11 displays a vertical cross sectional view of the pin. Other specifications such as material densities for the fuel materials are listed in Table 3.

Figure 10. Horizontal cross sectional view of pin cell model with dimensions (not to scale)

Graphite

Coolant, $r_{channel} = 5.3$ cm

Pitch = 20.3 cm

Fuel, $r_{fo} = 1.4605$ cm

Clad, $r_{co} = 1.61925$ cm
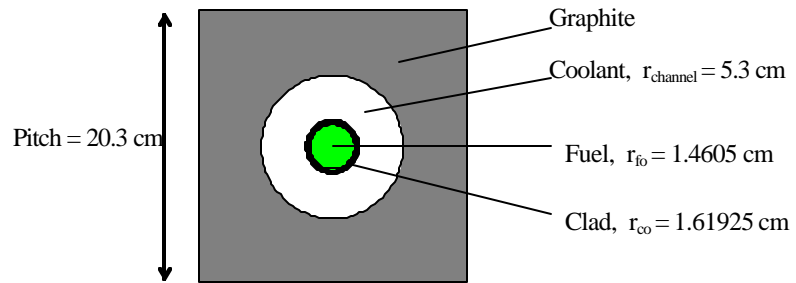
Figure 11. Vertical cross sectional view of fuel pin with dimensions (not to scale)

2.1167 cm

Graphite

Coolant

Cladding

101.6 cm

Fuel

2.1167 cm
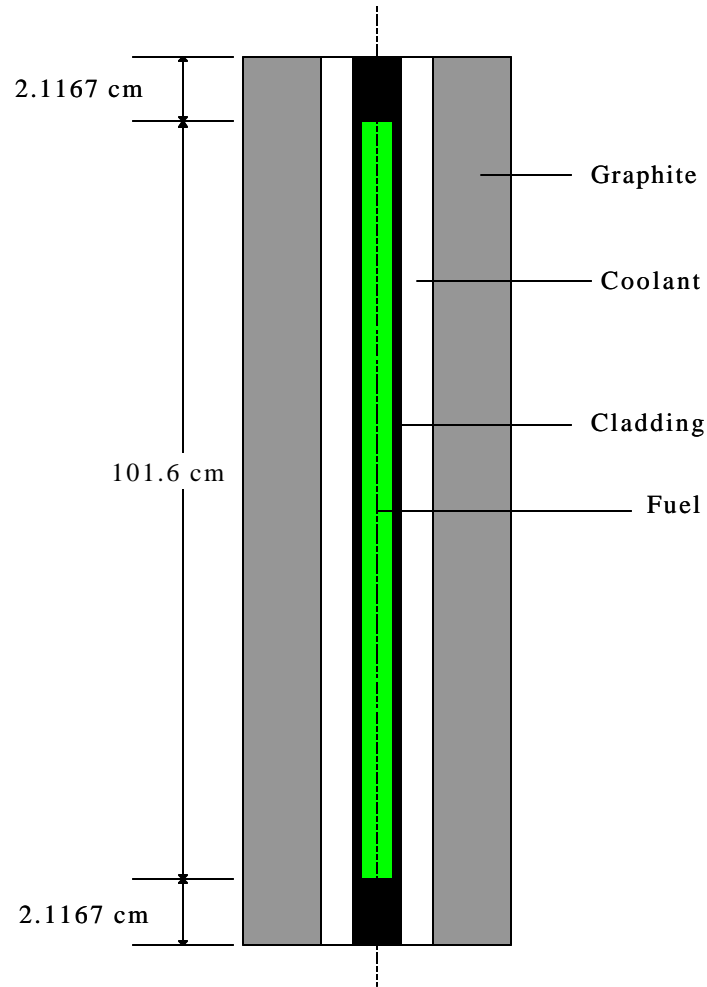
Grade A UK graphite with impurities was used for the moderator. A list of the impurities in the graphite and concentrations for each element are in Table 4. For the Monteburns simulation, the fuel cell geometry above was written to initially use the "candunau.lib" ORIGEN cross section library. This library, with information for a CANDU reactor with natural uranium metal fuel, was chosen since it contained initial information most closely related to the Calder Hall reactor, which was most specifically the natural uranium metal fuel. In subsequent burn steps Monteburns creates the ORIGEN library values using the data generated in MCNP for the isotopes which are included in the Monteburns input deck. Cross section information for other isotopes not included in the Monteburns input deck is taken from the initial ORIGEN library which was specified.

Table 4. Concentration of impurities in Grade A UK graphite

Element	Concentration [ppm]
Aluminum	7.0
Boron	0.016
Barium	10.0
Calcium	80.0
Chlorine	1.0
Chromium	2.5
Iron	25.0
Lithium	0.36
Nickel	6.0
Sulfur	50.0
Silicon	80.0
Titanium	8.0
Vanadium	40.0
Zinc	0.4

Core average temperature values were used to analyze the cross section files for the graphite, coolant, cladding, and fuel. These average temperatures are listed in Table 5. Cross section data files are available from the National Nuclear Data Center (NNDC) webpage maintained by Brookhaven National Laboratory and the Nuclear Information Service webpage maintained by Los Alamos National Laboratory.[21-22] The NNDC has four neutron cross section data libraries available. These include two versions of the Evaluated Nuclear Data Files (ENDF) from the USA, the Joint Evaluated Fission and Fusion (JEFF) library from Europe, and the Japanese Evaluated Nuclear Data Library (JENDL) from Japan.

The original pointwise cross section files for the pin cell materials from these libraries contain cross section data for each element or isotope at 300 degrees Kelvin. These files were processed using the computer code NJOY to the desired core average temperature for use with the Monteburns code.

Table 5. Core average temperatures for the pin cell model

Material	Temperature [K]
Fuel	695
Cladding	642
Coolant	511
Graphite Moderator	520

Using the average core specific power of 1.38 W/g, fuel density, and fuel volume, the power generated in the single fuel pin was determined to be 0.01706 MW. A total of 57 burn steps were used in Monteburns with the first two steps equaling one day and all subsequent steps being 80 days in length. Each 80-day time period at the average

power above corresponds to a burn step of 110.4 MWD/MTU. Total burnup of the fuel pin after completion of the 57 burn steps, or 4402 days, was 6074.76 MWD/MTU. Average core burnup for MAGNOX-style reactors is 3000 to 4000 MWD/MTU, but some cores can have a life of up to 6000 MWD/MTU before being discharged.[6] Thus, the range chosen here covers all possible burnup ranges for MAGNOX cores.

The simulation utilized a parallelized version of the Monteburns code with MCNP5 and ORIGEN 2.2. Each time the MCNP code was executed within the Monteburns simulation a total of 1.5 million particles were run (3000 particles in 500 kcode cycles). This calculation was executed on fourteen processors and took approximately 57.5 hours to complete. The Monteburns input files used are in Appendix A.

ORIGEN Library Verification

The ORIGEN cross section library for a fuel burnup of 3093 MWD/MTU was chosen to be the standard library for MAGNOX-style reactors. This library was extracted for an average burnup from step 30 of the Monteburns output. During the execution of Monteburns, a separate ORIGEN library file is created for each material. The files for the fuel and graphite were combined into one for use with the ORIGEN 2.2 code. The ORIGEN input deck includes one metric ton of natural uranium and one kilogram of the Grade A UK graphite for the fuel and moderator, respectively. The ORIGEN code does not have the ability to account for geometry, and since isotope concentration output values have units of grams of isotope per unit mass of initial uranium, it is advantageous to use one metric ton of uranium as the initial fuel inventory. A constant power irradiation with a 1.38 W/g average core power density was used for the fuel, and the graphite was processed using a constant flux irradiation. The neutron flux values in the graphite material at each burn step were obtained from the Monteburns output file for use in the ORIGEN flux irradiation of the graphite. These values are displayed in Appendix B with the ORIGEN input deck. This deck was used for

validation of the input cross section and fission product yield library that was produced for MAGNOX-style reactors.

Using the 'tape9' cross section library taken for an average burnup of approximately 3000 MWD/MTU, the ORIGEN input deck with one metric ton of natural uranium metal fuel and one kilogram of graphite was irradiated to an average burnup and beyond to determine the applicability of the cross section library at higher burnups. Table 6 displays the uranium and plutonium ORIGEN results as compared to the Monteburns results for each corresponding burnup. The percent difference values are calculated using:

$$\% \, Difference = \left[\frac{ORIGEN[\,g/MTU\,] - Monteburns[\,g/MTU\,]}{Monteburns[\,g/MTU\,]} \right] \times 100 \qquad (12)$$

and are displayed graphically in Figure 12.

At the average burnup, the cross section set performs well, with U-235 and Pu-239 having less than 2% and 1% error, respectively. The error increases for higher actinides at this burnup; however, larger errors are expected for these actinides.

Figure 12. Comparison of ORIGEN and Monteburns results in the fuel at various fuel burnups

Table 6. ORIGEN results using the new cross section library compared to Monteburns results at average and above average burnups for the fuel

Nuclide	Burnup= 3093 MWD/MTIHM			Burnup= 4087 MWD/MTIHM		
	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference
U-234	4.80E+01	4.81E+01	0.19%	4.62E+01	4.64E+01	0.34%
U-235	4.37E+03	4.44E+03	1.62%	3.75E+03	3.83E+03	2.12%
U-238	9.90E+05	9.90E+05	0.00%	9.89E+05	9.89E+05	0.00%
Pu-238	1.29E+00	1.20E+00	-7.31%	2.92E+00	2.62E+00	-10.28%
Pu-239	1.97E+03	1.96E+03	-0.36%	2.30E+03	2.28E+03	-0.98%
Pu-240	2.77E+02	2.72E+02	-1.71%	4.20E+02	4.07E+02	-3.05%
Pu-241	4.72E+01	4.13E+01	-12.44%	7.80E+01	7.60E+01	-2.58%
Pu-242	4.29E+00	3.30E+00	-23.12%	9.86E+00	8.32E+00	-15.55%
Nuclide	Burnup= 5081 MWD/MTIHM			Burnup= 6074 MWD/MTIHM		
	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference
U-234	4.45E+01	4.47E+01	0.52%	4.29E+01	4.32E+01	0.70%
U-235	3.21E+03	3.30E+03	2.68%	2.75E+03	2.84E+03	3.24%
U-238	9.88E+05	9.88E+05	0.01%	9.86E+05	9.87E+05	0.01%
Pu-238	5.63E+00	5.08E+00	-9.82%	9.63E+00	8.89E+00	-7.63%
Pu-239	2.56E+03	2.52E+03	-1.52%	2.76E+03	2.70E+03	-2.05%
Pu-240	5.71E+02	5.43E+02	-4.93%	7.27E+02	6.76E+02	-7.12%
Pu-241	1.12E+02	1.18E+02	5.38%	1.47E+02	1.65E+02	11.81%
Pu-242	1.83E+01	1.67E+01	-8.82%	2.98E+01	2.89E+01	-3.02%

To examine the source of increased error for higher actinides, Figure 2 in Chapter I displays the process by which plutonium is produced in a reactor during operation. For each isotope, a balance equation tracks the concentration of these isotopes where the rate of change is equivalent to the production minus loss rates. For each higher actinide, production of the isotope depends on a loss mechanism of a lower actinide. One example is the production of Pu-239 from the beta decay of Np-239. Np-239 is produced from the beta decay of U-239 which is produced from the radiative capture of a neutron in U-238. Each calculation for an isotope has some related error. As these calculations proceed up through the higher actinides, these errors increase as errors from previous

calculations are propagated through each subsequent calculation. Additionally, the amount of material produced for each isotope decreases for each higher actinide because of the physics and means through which they are created. For these reasons, the concentrations of higher actinides are less accurate than actinides close to the mass of the original materials.

For increasing burnups, the accuracy of the U-235, Pu-239 and Pu-240 concentrations decreases. For the highest burnup, the error between Monteburns and ORIGEN (using the new library) for the U-235, Pu-239 and Pu-240 concentrations is just over 3%, 2%, and 7%, respectively. However, for Pu-238, Pu-241, and Pu-242 the concentrations do not follow the same pattern. Pu-238 values at nearly double the average burnup have approximately the same accuracy as the results at the average burnup. Pu-241 is most accurate at 4087 and 5081 MWD/MTU, and Pu-242 values are increasingly more accurate as the burnup is increased.

A similar comparison was done for the isotopes in the graphite. For elements where a cross section for the natural element was used in the Monteburns deck—carbon, chlorine, calcium, sulfur, titanium, vanadium, and zinc—no data is available to compare for the corresponding isotopes of each element from Monteburns. During the execution of Monteburns, isotopes for which data is desired must be listed in the Monteburns input file. However, isotopes that are listed in this file must also be included in the associated MCNP file and must have cross section files to be included. For the isotopes where natural elemental cross section files are used in the initial material declarations, cross section files for each individual isotope were not available, or the files which were available were unable to be processed with NJOY. However, these isotopes were still tracked within the Monteburns code, but no mechanism is available for the user to extract this data. The isotopes which did have data from Monteburns are listed in Table 7 and Table 8. Table 7 has graphite data for average burnups in the fuel and Table 8 for above average burnups.

Table 7. ORIGEN results using the new cross section library compared to Monteburns results in the graphite for average burnups

Nuclide	Burnup= 3093 MWD/MTIHM			Burnup= 4087 MWD/MTIHM		
	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference
Li-6	6.78E-06	6.94E-06	2.33%	4.57E-06	4.66E-06	2.08%
Li-7	3.43E-04	3.43E-04	0.00%	3.43E-04	3.43E-04	0.00%
Be-9	1.65E-04	1.65E-04	-0.11%	2.19E-04	2.19E-04	-0.12%
B-10	1.89E-08	2.07E-08	9.58%	3.77E-09	4.09E-09	8.51%
B-11	1.30E-05	1.31E-05	0.02%	1.30E-05	1.31E-05	0.02%
Al-27	7.00E-03	7.00E-03	-0.03%	7.00E-03	7.00E-03	-0.04%
Si-28	7.35E-02	7.35E-02	0.00%	7.35E-02	7.35E-02	-0.02%
Si-29	3.87E-03	3.87E-03	-0.03%	3.88E-03	3.87E-03	-0.04%
Si-30	2.65E-03	2.65E-03	0.00%	2.65E-03	2.65E-03	0.00%
Cr-50	1.02E-03	1.02E-03	0.05%	1.01E-03	1.02E-03	0.05%
Cr-52	2.11E-02	2.09E-02	-0.91%	2.11E-02	2.09E-02	-1.22%
Cr-53	2.38E-03	2.38E-03	0.06%	2.37E-03	2.37E-03	0.04%
Cr-54	6.71E-04	6.70E-04	-0.15%	6.89E-04	6.88E-04	-0.13%
Fe-54	1.41E-03	1.41E-03	0.09%	1.40E-03	1.41E-03	0.13%
Fe-56	2.29E-02	2.29E-02	0.01%	2.29E-02	2.29E-02	0.04%
Fe-57	6.18E-04	6.17E-04	-0.23%	6.43E-04	6.42E-04	-0.20%
Fe-58	7.49E-05	7.49E-05	-0.05%	7.56E-05	7.56E-05	-0.04%
Ni-58	4.01E-03	4.01E-03	0.04%	4.00E-03	4.00E-03	0.04%
Ni-60	1.60E-03	1.60E-03	-0.02%	1.60E-03	1.60E-03	-0.02%
Ni-61	7.67E-05	7.66E-05	-0.13%	7.85E-05	7.84E-05	-0.11%
Ni-62	2.26E-04	2.26E-04	0.04%	2.25E-04	2.25E-04	0.02%
Ni-64	6.04E-05	6.04E-05	0.01%	6.04E-05	6.04E-05	0.01%
Ba-130	9.71E-06	9.71E-06	-0.04%	9.61E-06	9.61E-06	-0.03%
Ba-132	9.67E-06	9.67E-06	-0.02%	9.66E-06	9.66E-06	-0.01%
Ba-134	2.35E-04	2.35E-04	-0.11%	2.35E-04	2.34E-04	-0.15%
Ba-135	6.37E-04	6.37E-04	0.00%	6.34E-04	6.34E-04	0.00%
Ba-136	7.87E-04	7.88E-04	0.02%	7.91E-04	7.91E-04	0.02%
Ba-137	1.11E-03	1.11E-03	0.00%	1.11E-03	1.11E-03	0.03%
Ba-138	7.20E-03	7.20E-03	-0.03%	7.21E-03	7.20E-03	-0.04%

The data in Table 7 and Table 8 show that the ORIGEN library performs well at all burnups when doing a flux irradiation of the graphite. During the execution of Monteburns, the code outputs the average neutron flux in each material for each burn step. These values were taken and used for equivalent burn steps in a flux irradiation calculation of the graphite in ORIGEN 2.2 with the new cross section library. Only one

isotope, B-10, has a significant percent difference between the ORIGEN and Monteburns calculations, and the majority of isotopes vary from the Monteburns results by less than one percent for all burnups. Overall, the ORIGEN library that was developed for MAGNOX-style reactors is successful for use at average burnups of graphite-moderated reactors.
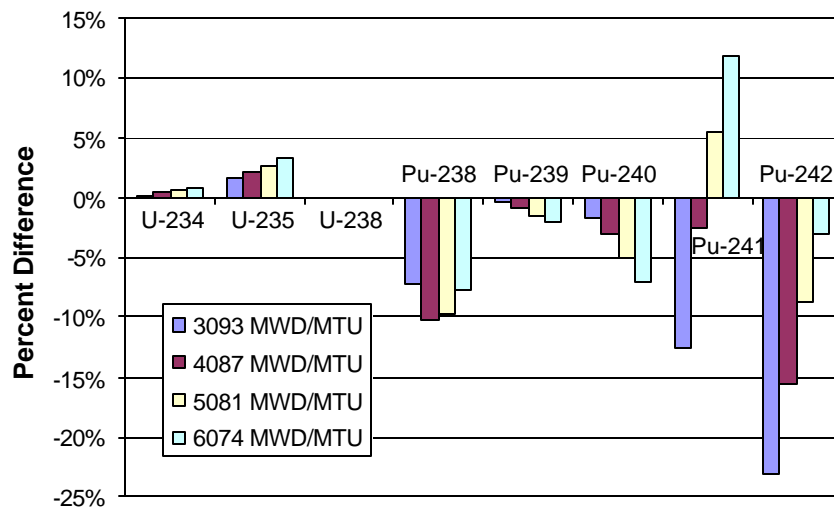
Table 8. ORIGEN results using the new cross section library compared to Monteburns results in the graphite for above average burnups

Nuclide	Burnup= 5081 MWD/MTIHM			Burnup= 6074 MWD/MTIHM		
	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference	Monteburns [g/MTIHM]	ORIGEN [g/MTIHM]	% Difference
Li-6	3.07E-06	3.12E-06	1.63%	2.05E-06	2.08E-06	1.03%
Li-7	3.43E-04	3.43E-04	0.00%	3.43E-04	3.43E-04	0.00%
Be-9	2.73E-04	2.73E-04	-0.08%	3.29E-04	3.28E-04	-0.10%
B-10	7.47E-10	7.95E-10	6.47%	1.48E-10	1.54E-10	3.88%
B-11	1.30E-05	1.31E-05	0.02%	1.30E-05	1.31E-05	0.02%
Al-27	7.00E-03	6.99E-03	-0.05%	7.00E-03	6.99E-03	-0.05%
Si-28	7.35E-02	7.35E-02	-0.02%	7.35E-02	7.35E-02	-0.03%
Si-29	3.88E-03	3.88E-03	-0.02%	3.89E-03	3.89E-03	-0.03%
Si-30	2.65E-03	2.65E-03	0.00%	2.65E-03	2.65E-03	0.00%
Cr-50	1.01E-03	1.01E-03	0.04%	1.00E-03	1.00E-03	0.04%
Cr-52	2.12E-02	2.09E-02	-1.49%	2.13E-02	2.09E-02	-1.79%
Cr-53	2.36E-03	2.36E-03	0.06%	2.35E-03	2.35E-03	0.03%
Cr-54	7.08E-04	7.07E-04	-0.10%	7.26E-04	7.26E-04	-0.05%
Fe-54	1.40E-03	1.40E-03	0.11%	1.40E-03	1.40E-03	0.16%
Fe-56	2.28E-02	2.28E-02	0.02%	2.28E-02	2.28E-02	0.05%
Fe-57	6.68E-04	6.67E-04	-0.15%	6.93E-04	6.92E-04	-0.09%
Fe-58	7.63E-05	7.62E-05	-0.04%	7.70E-05	7.69E-05	-0.02%
Ni-58	3.99E-03	3.99E-03	0.04%	3.98E-03	3.98E-03	0.04%
Ni-60	1.60E-03	1.60E-03	0.03%	1.60E-03	1.60E-03	0.03%
Ni-61	8.03E-05	8.03E-05	-0.09%	8.22E-05	8.21E-05	-0.05%
Ni-62	2.24E-04	2.24E-04	-0.01%	2.22E-04	2.22E-04	-0.03%
Ni-64	6.04E-05	6.04E-05	0.00%	6.04E-05	6.05E-05	0.01%
Ba-130	9.51E-06	9.51E-06	-0.04%	9.41E-06	9.41E-06	-0.04%
Ba-132	9.65E-06	9.65E-06	-0.01%	9.64E-06	9.64E-06	0.00%
Ba-134	2.35E-04	2.34E-04	-0.19%	2.34E-04	2.34E-04	-0.21%
Ba-135	6.31E-04	6.31E-04	0.01%	6.27E-04	6.27E-04	0.01%
Ba-136	7.94E-04	7.94E-04	0.02%	7.98E-04	7.98E-04	0.02%
Ba-137	1.11E-03	1.11E-03	-0.03%	1.10E-03	1.11E-03	0.00%
Ba-138	7.21E-03	7.21E-03	-0.05%	7.21E-03	7.21E-03	-0.07%

Figure 13 graphically displays a comparison of the percent differences for the three isotopes in the graphite where the difference between the ORIGEN and Monteburns results were greater than one percent: Li-6, B-10, and Cr-52. Two of these isotopes, Li-6 and B-10, are indicator isotopes in GIRM.

Figure 13. Comparison of ORIGEN and Monteburns results in the graphite for various fuel burnups
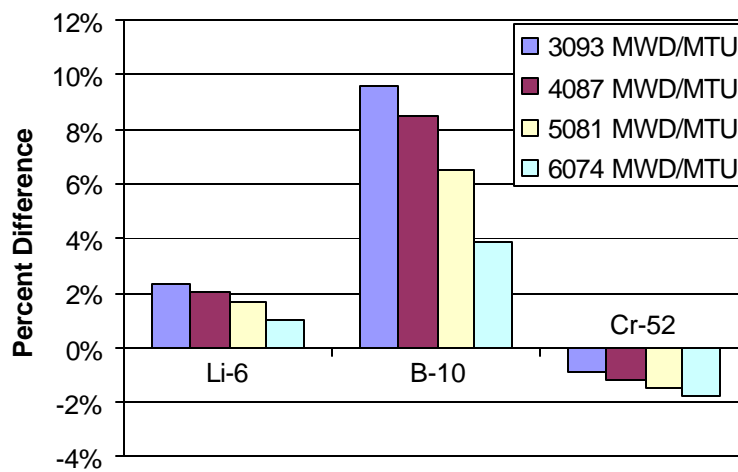
CHAPTER III

LOCAL PLUTONIUM ESTIMATES WITH

MONTEBURNS

During the analysis portion of GIRM, reactor physics calculations lead to the development of correlations to relate the plutonium production in the fuel to specific values of isotope ratios in the graphite. Previous analyses using data from the Trawsfynydd Unit 2 reactor developed such correlations for the plutonium production estimate as a function of the Ti-48/Ti-49 ratio.[23] Using results from the Monteburns pin cell simulation which was utilized in the development of the ORIGEN cross section library, similar correlations for other isotope values to the neutron fluence in the reactor produced local plutonium estimates for this case. Figure 14 displays the production of plutonium in the Calder Hall fuel pin with increasing fuel burnup. The plutonium values in the figure are reported in grams of plutonium—which is the sum of the Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242 isotopes—per centimeter of active fuel length. These plutonium production estimates are considered local estimates for a corresponding graphite sample location in the GIRM methodology.

The data in Figure 14 was fit with a trendline using a standard regression technique. The equation represents the relationship between the total plutonium produced in the fuel pin as a function of the burnup in the fuel. Thus, if the burnup in the fuel pin is known, the total plutonium produced in the pin can be determined from the equation.

Figure 14. Plutonium production in the Calder Hall pin cell simulation as a function of fuel burnup
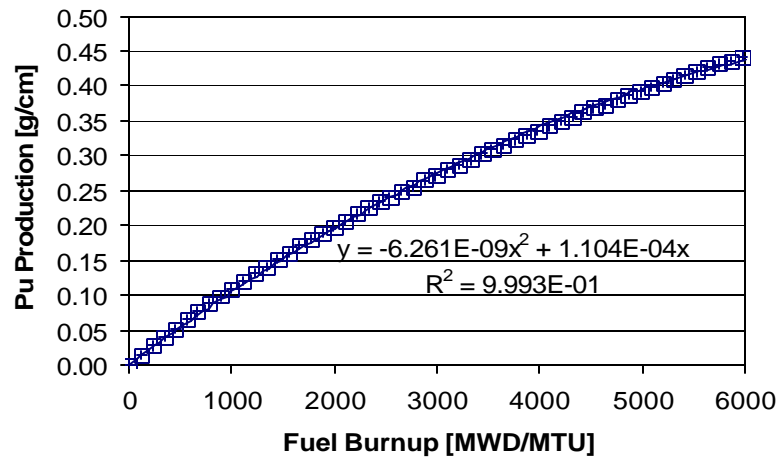
Similar to the Trawsfynydd analysis, the output from the Monteburns computation was used to determine correlations for the isotope ratios of Li-6/Li-7 and B-10/B-11 which are both used with the GIRM method. The Li-6/Li-7 ratio is applicable for low to intermediate fluence ranges, and the B-10/B-11 ratio is applicable for low fluence ranges. Figure 15 displays the changes in Li-6 and Li-7 concentrations in the graphite moderator as a function of burnup in the fuel pin. Figure 16 shows the correlation between the boron isotope concentrations in the moderator and the burnup in the fuel pin.

Figure 15. Lithium isotope concentrations with respect to fuel burnup from Calder Hall pin cell calculation
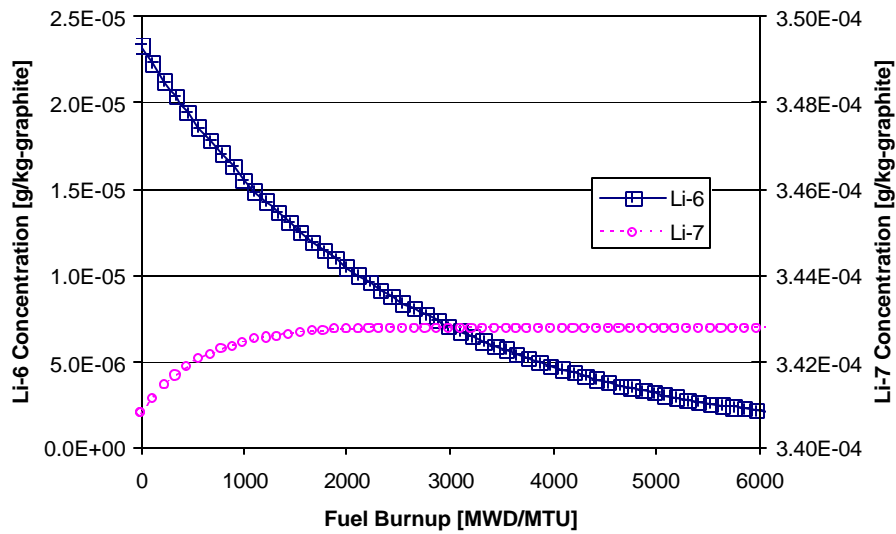
Figure 16. Boron isotope concentrations as a function of fuel burnup from the Calder Hall pin cell simulation
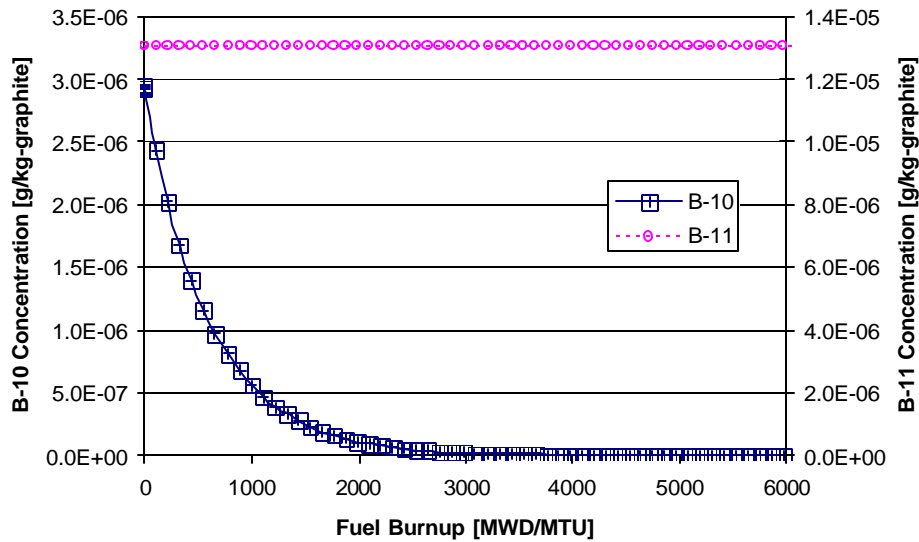
Taking ratios of these isotopes, Figure 17 displays the isotopic ratios for each element as a function of burnup in the Calder Hall pin cell model. A trend line using a standard regression analysis was fit to the data with the corresponding equations shown on the graph. These equations for the curves with x representing the value of fuel burnup in units of [MWD/MTU] are:

$$^{10}B / {}^{11}B = 2.202E\text{-}01 e^{-1.625E\text{-}03x} \tag{13}$$

$$^{6}Li / {}^{7}Li = 6.802E\text{-}02 e^{-3.993E\text{-}04x} \tag{14}$$

The behavior of the ratios in Figure 17 also explains the fluence ranges for which each ratio is applicable. The interval over which the ratio has the greatest changes in value is the suitable range for use as an indicator in GIRM. Intervals where the ratio stays nearly constant provide little information on the exact state of the reactor in inverse calculations such as GIRM. Thus, the B-10/B-11 ratio is essentially useful only for burnup ranges less than 2000 MWD/MTU.

Figure 17. Lithium and boron ratios as a function of fuel burnup from the Calder Hall pin cell simulation
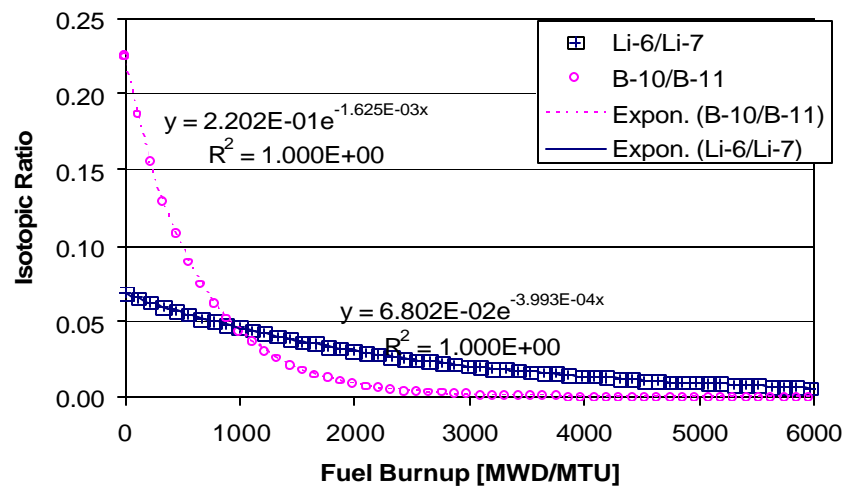
To obtain a direct relationship between the isotope ratio and the plutonium production, the data was examined in the applicable range for each ratio. Low fluence ranges for the boron ratio was taken to be from zero to 2000 MWD/MTU. The value for the B-10/B-11 ratio at each fuel burnup was plotted with the corresponding plutonium production value for that. The low to intermediate fluence ranges for the lithium ratio was taken to be from zero to approximately 4000 MWD/MTU. Similarly, the Li-6/Li-7 ratio value was plotted with the corresponding plutonium value up to a fuel burnup of 4000 MWD/MTU. The result is Figure 18 below.

Figure 18. Plutonium produced in the Calder Hall fuel pin as a function of lithium and boron isotope ratios
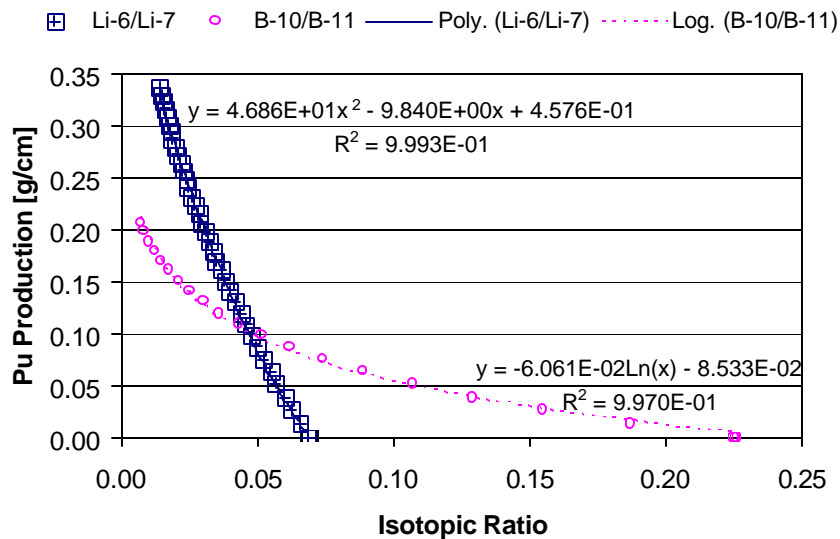
Figure 18 graphically displays the direct correlation between the lithium and boron isotope ratios and the plutonium production in the fuel pin for the Calder Hall pin. This data comes from the reactor physics calculations using the earlier described pin cell

model in Monteburns. The equation for each trend line is shown on the graph. These equations represent an example of the methodology for a local plutonium estimate in the GIRM technique. This local estimate is associated with the corresponding graphite sample taken from a location that is representative of the pin cell model calculation in Monteburns. For each local estimate there are related sources of error. Some of these sources of error include uncertainty in the graphite sample measurements, calculational methods used, fuel parameter specifications, and declared or undeclared parameters regarding the operational history of the reactor.

Variation of Global Plutonium Estimate

The GIRM method global plutonium estimate involves the development of a three-dimensional, full-core fluence profile. The fluence profile is derived with a regression analysis from the set of local energy production estimates. This method for a global plutonium estimate is advantageous in its ability to account for variations in control rod positions and other uncertainties associated with the reactor's operating history.[24]

There is an alternative procedure for estimating the plutonium produced in the entire reactor core which does not require the development of a regression model for the core fluence. This method would make use of a full-core simulation of the reactor to determine an estimate of the radial and axial power (or fluence) profile throughout the core. Local plutonium estimates would be calculated using the same method described above. These local estimates are then used to normalize the full-core fluence shape to an average burnup for the whole core. Each local estimate could then be treated as an independent measure of the full-core burnup (for example, Figure 19 displays a hypothetical distribution of full-core burnup estimates calculated from a set of graphite samples). The statistical deviation of the sample points provides an average and intrinsic statistical uncertainty in the entire approach. The mean of these hypothetical samples is 1026 MWD/MTU with a standard deviation of 57 MWD/MTU.

Figure 19. Hypothetical distribution of burnup from GIRM samples
(mean = 1026 MWD/MTU, standard deviation = 57 MWD/MTU)
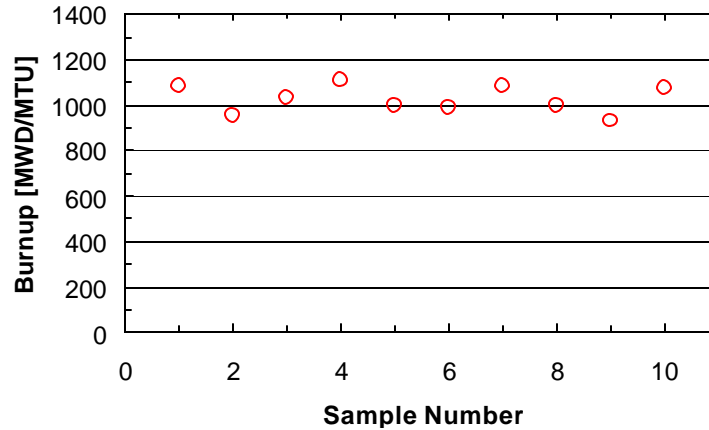
Following the calculation of the average core burnup, the total plutonium produced from the core could be determined by using the estimate of plutonium versus burnup from the pin cell calculation (see Figure 20). Since this calculation is for an average fuel pin and since the burnup is the fuel pin average, this plutonium production estimate would be the average plutonium produced per unit mass of fuel. Figure 20 displays the relation of the plutonium inventory for the reactor core as a function of average core burnup for an example reactor. The red line represents the sum of all plutonium isotopes found in the fuel rods, and the blue line is the inventory of Pu-240 from all fuel rods (Pu-240 is the dominant producer of spontaneous fission neutrons from the fuel and is the primary isotope used when measuring plutonium in spent fuel). Taking the average burnup value from the hypothetical results above in Figure 19— approximately 1000 MWD/MTU with the associated statistical deviation—the total production of plutonium is found to be approximately 110 kg as found in Figure 20. Standard error propagation of the statistical deviation above results in a measure of the error in the total plutonium estimate using this alternate method. The disadvantage to

this approach compared to the GIRM regression analysis is that it requires detailed full-core design information and operational history of the reactor.

Figure 20. Correlation of plutonium production to the average core burnup from a full core reactor physics model
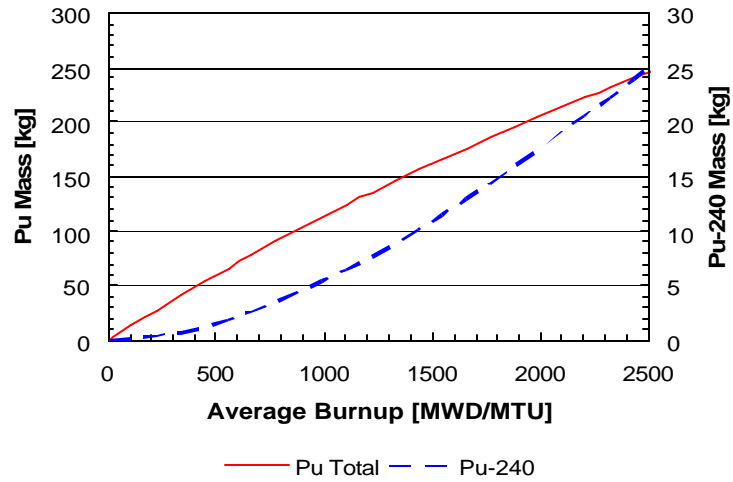
CHAPTER IV

SENSITIVITY ANALYSIS

For use in safeguards analysis and materials accountability, every detail of the operating history of the reactor or every parameter of the material will not be perfectly known. Also, since safeguards methods are applied to a country that could possibly be in violation of its safeguards responsibilities, some information provided by the state may not be reliable. Therefore, it is important to know which factors and parameters have the greatest impact on the answers determined by GIRM and other similar techniques. This information is most often obtained from thorough sensitivity analyses of the methods used.

To demonstrate the use of the ORIGEN library for MAGNOX-style reactors in a sensitivity analysis, various parameters of the ORIGEN input deck were changed. This analysis focuses on the uncertainties in material composition and material properties and the magnitude these uncertainties cause in the results. Previous studies of the GIRM method have analyzed the total error associated with the regression method and found that these errors are small especially when many samples are used.[12,14,24]

During the application of GIRM for a comparative plutonium estimate in the Trawsfynydd reactor, the analysis of graphite samples concluded that the titanium impurities in the graphite had a substantial heterogeneity throughout the matrix. Additionally, a significant amount of variance in the titanium ratios was seen beyond the expected errors.[24] The result of the analysis was a global plutonium production estimate for the Trawsfynydd reactor with an associated error dependent on the regression model that was used. Another error analysis evaluated the contributions associated with the individual mass spectroscopic measurements, the optimization of sampling methodology (including location and number of samples), and local reactor physics error to the uncertainty in the total plutonium estimate.[14] The final conclusion was a root mean square error of 1.62% for the GIRM plutonium estimate. However, these analyses did not evaluate the contributions from all uncertainties which can be present in the material

properties, namely the concentration of trace impurities. It is expected that the dominant source of error for the GIRM method is likely to be in the input parameters (material concentrations and cross sections) used in the pin cell model.

A sensitivity analysis is useful in finding the specific parametric sources of error and to quantify the level of error that uncertainty in these specific parameters can cause in the results of GIRM. Using the library developed for the ORIGEN code, several cases were run to analyze the effect of small changes in the selected input parameters. The first two parametric studies focus on uncertainties regarding information for the fuel. The last analysis examines variations in the graphite. These analyses are:

- The effect of the initial weight percent of aluminum alloy in the fuel on the uranium and plutonium isotopic ratios at the end of burnup
- The change in uranium and plutonium isotopic ratios at the end of burnup due to $\pm 1\%$ and $\pm 5\%$ changes in the uranium and plutonium cross section values used in the calculations
- The dependence at a specified burnup of the isotopic ratios of trace elements in the graphite to each element's initial concentration

The ORIGEN code has both advantages and disadvantages in its use for a sensitivity analysis. The most important advantage to the code's use in a sensitivity analysis is the speed at which it operates. A single simulation executes on the order of seconds. This accommodates numerous variations in the input parameters. In addition to its rapid execution, the cross section libraries used with ORIGEN are easily interpreted text documents. It is an easy task for the user to make changes in the cross section library to analyze fluctuations and uncertainties associated with the result of the GIRM method due to the cross section values.

Other transport codes, such as MCNP or Monteburns, can be extremely lengthy in execution. For instance, the Monteburns pin cell simulation that was described in Chapter II took over 57 hours to finish with fourteen processors running in parallel to complete the calculation. With each additional burn step or added complication to the geometry, the computational time increases significantly. However, each ORIGEN

simulation run in this sensitivity analysis took only seconds to compute. The cross section files used for MCNP and Monteburns are also large, cumbersome files with point-wise data for each isotope or element. This is an impractical format for manual changes. On the other hand, the ORIGEN code does not possess the capability to analyze geometric considerations in the system. Variations from the location of the graphite sample—such as the axial height, radial position, and depth drilled into the graphite block—cannot be assessed. Considerations such as these require the use of Monteburns, WIMS, or other lattice physics codes.[8,17]

Variations in Aluminum Alloy Concentration

Most uranium metal fuels are an alloy, and uranium is often alloyed with small percentages of aluminum. In this analysis, the aluminum alloy percentage was altered to simulate uncertainty in the initial fuel composition. Table 9 displays the amount of grams per metric ton of initial uranium at 3093 MWD/MTU using different aluminum alloy concentrations at the beginning of life. The values under each column labeled "ORIGEN [g/MTU]" are the grams of each corresponding isotope per metric ton of initial uranium that is present in the fuel following a constant power irradiation at 1.38 W/gU to a burnup of 3093MWD/MTU. The difference percentages use the 0.0 w/o Al data as the basis for comparison with Equation 11 where '#' is either 0.5, 1.0, or 2.0.

$$\%Difference = \left[\frac{ORIGEN[\,g/MTU\,]_{\#w/oAl} - ORIGEN[\,g/MTU\,]_{0.0w/oAl}}{ORIGEN[\,g/MTU\,]_{0.0w/oAl}} \right] \times 100 \qquad (15)$$

The change in the uranium results is approximately the same as the percentage of initial aluminum in the fuel. For 1.0 w/o (percent by weight) initial aluminum alloy, the initial uranium concentration is 1.0 w/o less than without aluminum. Pu-239 follows similar behavior but is less than the original result by approximately half the weight percent concentration of aluminum. For example, in the simulation with 1.0 w/o Al, the Pu-239 is 0.41% below the result without any alloy—approximately half the magnitude

of the change in aluminum alloy weight percent found initially in the fuel. The other plutonium isotopes, however, overestimate the concentrations as compared to the standard without aluminum. Figure 21 graphically displays the information from Table 9 for the set of uranium and plutonium isotopes.

Table 9. Comparison of uranium and plutonium results at a burnup of 3093 MWD/MTU for varying initial aluminum alloy concentrations in the fuel

Nuclide	0.0 w/o Al	0.5 w/o Al		1.0 w/o Al		2.0 w/o Al	
	ORIGEN [g/MTU]	ORIGEN [g/MTU]	% Difference	ORIGEN [g/MTU]	% Difference	ORIGEN [g/MTU]	% Difference
U-234	4.807E+01	4.781E+01	-0.54%	4.754E+01	-1.10%	4.700E+01	-2.23%
U-235	4.436E+03	4.403E+03	-0.74%	4.371E+03	-1.47%	4.306E+03	-2.93%
U-236	4.19E+02	4.18E+02	-0.1%	4.18E+02	-0.3%	4.17E+02	-0.5%
U-238	9.896E+05	9.846E+05	-0.51%	9.797E+05	-1.00%	9.697E+05	-2.01%
Pu-238	1.195E+00	1.203E+00	0.67%	1.212E+00	1.42%	1.228E+00	2.76%
Pu-239	1.960E+03	1.956E+03	-0.20%	1.952E+03	-0.41%	1.944E+03	-0.82%
Pu-240	2.723E+02	2.729E+02	0.22%	2.736E+02	0.48%	2.750E+02	0.99%
Pu-241	4.131E+01	4.159E+01	0.68%	4.187E+01	1.36%	4.244E+01	2.74%
Pu-242	3.301E+00	3.341E+00	1.21%	3.381E+00	2.42%	3.465E+00	4.97%

Figure 21. Difference of ORIGEN results compared to case with no aluminum alloy in fuel
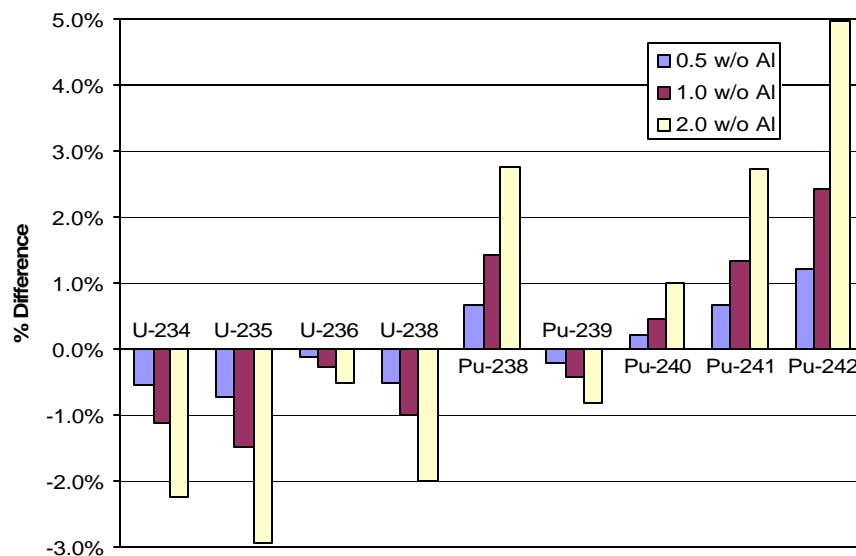
Uranium Cross Section Variations

  The sensitivity of the uranium cross section data was next analyzed. Neutron cross sections are measured quantities and thus have an associated error. Analyzing fluctuations in the U-235 and U-238 neutron cross section values can help quantify the magnitude of the effects from this uncertainty.

  Both the U-235 and U-238 cross section data in the ORIGEN library were changed by ±1% and ±5% for a total of ten different scenarios. The results of these cases determine the effect of the cross section fluctuations on the uranium and plutonium measurements at the end of irradiation. The results for these parameter changes are shown in Table 10 and Table 11. Table 10 displays the results at the end of irradiation when the input cross section data for U-235 was changed. Table 11 displays the results at the end of irradiation when the input cross section data for U-238 was changed. In the tables, "XS" is used to represent "cross-section", and the percent difference values are calculated using:

$$Error = \left[\frac{(\pm\%)[g/MTU] - Original[g/MTU]}{Original[g/MTU]} \right] \times 100 \tag{16}$$

  The change in answers for the uranium isotopes is approximately 0.5% or less for all cross section variations. The Pu-239 and Pu-240 results were also little affected by the change in parameter data (less than 1%). Pu-238, Pu-241, and Pu-242 results were affected more greatly by the change in cross section data but the results still did not change by more than 2%. With the exception of the Pu-242/Pu-239 ratio, the change in all isotope ratios was less than 1%. The difficulty in accurately calculating the higher actinides is another contributing factor in the larger variance of results for the higher plutonium isotopes.

Table 10. Comparison of ORIGEN results at a burnup of 3093 MWD/MTU for various changes in the U-235 cross section

Nuclide	Original	+5% U-235 XS Change		+1% U-235 XS Change		-1% U-235 XS Change		-5% U-235 XS Change	
	[g/MTU]	[g/MTU]	Error	[g/MTU]	Error	[g/MTU]	Error	[g/MTU]	Error
U-234	4.81E+01	4.82E+01	0.31%	4.81E+01	0.06%	4.80E+01	-0.06%	4.79E+01	-0.31%
U-235	4.44E+03	4.39E+03	-1.08%	4.43E+03	-0.23%	4.45E+03	0.20%	4.49E+03	1.13%
U-236	4.19E+02	4.26E+02	1.79%	4.20E+02	0.36%	4.17E+02	-0.38%	4.11E+02	-1.93%
U-238	9.90E+05	9.90E+05	0.01%	9.90E+05	0.00%	9.90E+05	0.00%	9.90E+05	-0.01%
Pu-238	1.20E+00	1.13E+00	-5.52%	1.18E+00	-1.17%	1.21E+00	1.26%	1.27E+00	6.11%
Pu-239	1.96E+03	1.93E+03	-1.58%	1.95E+03	-0.31%	1.97E+03	0.31%	1.99E+03	1.63%
Pu-240	2.72E+02	2.62E+02	-3.97%	2.70E+02	-0.84%	2.75E+02	0.81%	2.84E+02	4.19%
Pu-241	4.13E+01	3.88E+01	-6.05%	4.08E+01	-1.26%	4.19E+01	1.31%	4.40E+01	6.58%
Pu-242	3.30E+00	3.01E+00	-8.75%	3.24E+00	-1.88%	3.36E+00	1.91%	3.62E+00	9.78%
U-234/ U-238	4.86E-05	4.87E-05	0.30%	4.86E-05	0.06%	4.85E-05	-0.06%	4.84E-05	-0.30%
U-235/ U-238	4.48E-03	4.43E-03	-1.09%	4.47E-03	-0.23%	4.49E-03	0.20%	4.53E-03	1.14%
U-236/ U-238	4.23E-04	4.31E-04	1.78%	4.25E-04	0.36%	4.22E-04	-0.38%	4.15E-04	-1.92%
Pu-240/ Pu-239	1.39E-01	1.36E-01	-2.42%	1.38E-01	-0.54%	1.40E-01	0.50%	1.42E-01	2.51%
Pu-241/ Pu-239	2.11E-02	2.01E-02	-4.54%	2.09E-02	-0.96%	2.13E-02	1.00%	2.21E-02	4.87%
Pu-242/ Pu-239	1.68E-03	1.56E-03	-7.29%	1.66E-03	-1.58%	1.71E-03	1.60%	1.82E-03	8.02%

Table 11. Comparison of ORIGEN results at a burnup of 3093 MWD/MTU for various changes in the U-238 cross section

Nuclide	Original	+5% U-238 XS Change		+1% U-238 XS Change		-1% U-238 XS Change		-5% U-238 XS Change	
	[g/MTU]	[g/MTU]	Error	[g/MTU]	Error	[g/MTU]	Error	[g/MTU]	Error
U-234	4.81E+01	4.82E+01	0.19%	4.81E+01	0.04%	4.81E+01	-0.02%	4.80E+01	-0.17%
U-235	4.44E+03	4.47E+03	0.68%	4.44E+03	0.14%	4.43E+03	-0.16%	4.40E+03	-0.72%
U-236	4.19E+02	4.14E+02	-1.12%	4.18E+02	-0.24%	4.206E+02	0.21%	4.24E+02	1.15%
U-238	9.90E+05	9.90E+05	-0.01%	9.90E+05	0.00%	9.90E+05	0.00%	9.90E+05	0.01%
Pu-238	1.20E+00	1.20E+00	0.75%	1.20E+00	0.17%	1.19E+00	-0.08%	1.19E+00	-0.75%
Pu-239	1.96E+03	2.04E+03	4.13%	1.98E+03	0.82%	1.94E+03	-0.82%	1.88E+03	-4.13%
Pu-240	2.72E+02	2.80E+02	2.68%	2.74E+02	0.51%	2.71E+02	-0.55%	2.65E+02	-2.83%
Pu-241	4.13E+01	4.19E+01	1.43%	4.14E+01	0.29%	4.12E+01	-0.29%	4.07E+01	-1.50%
Pu-242	3.30E+00	3.30E+00	-0.15%	3.30E+00	-0.03%	3.30E+00	0.00%	3.30E+00	0.06%
U-234/ U-238	4.86E-05	4.87E-05	0.20%	4.86E-05	0.04%	4.86E-05	-0.02%	4.85E-05	-0.18%
U-235/ U-238	4.48E-03	4.51E-03	0.69%	4.49E-03	0.14%	4.48E-03	-0.16%	4.45E-03	-0.73%
U-236/ U-238	4.23E-04	4.18E-04	-1.11%	4.22E-04	-0.24%	4.24E-04	0.21%	4.28E-04	1.14%
Pu-240/ Pu-239	1.39E-01	1.37E-01	-1.39%	1.39E-01	-0.30%	1.39E-01	0.27%	1.41E-01	1.36%
Pu-241/ Pu-239	2.11E-02	2.05E-02	-2.60%	2.10E-02	-0.52%	2.12E-02	0.53%	2.17E-02	2.75%
Pu-242/ Pu-239	1.68E-03	1.62E-03	-4.11%	1.67E-03	-0.84%	1.70E-03	0.82%	1.76E-03	4.37%

Figure 22-Figure 25 graphically display the change in the concentration of each uranium isotope as a function of variation in the U-235 and U-238 cross section values. The calculated concentration of the uranium isotopes is linearly related to the value of the U-235 and U-238 cross sections used in the ORIGEN library. The equations displayed on each figure are the best fit straight line for the data. The R^2 value indicates how well the trend line fits the data, and the trend line is most reliable when the R^2 value is near one. The slope of the trend lines are indicative of the magnitude of the effect that the cross sections have on the calculated concentrations at the 3093 MWD/MTU burnup. This value is also useful for the propagation of error in the results of GIRM due to a quantitative uncertainty in the cross section.

Figure 22. Effect of uranium cross sections on the U-234 concentration at 3093 MWD/MTU
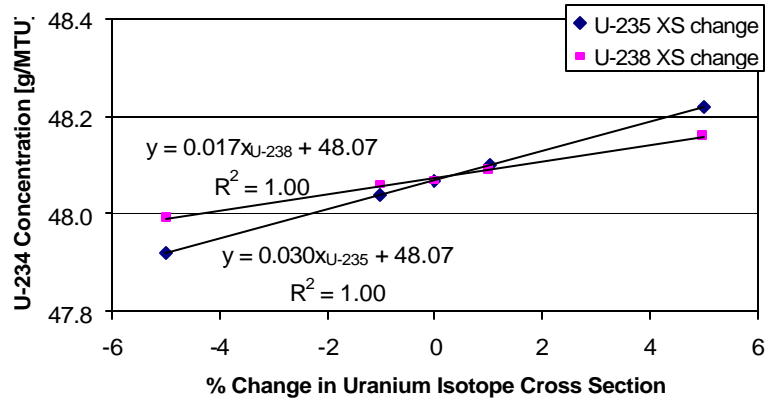
Figure 23. Effect of uranium cross sections on the U-235 concentration at 3093 MWD/MTU
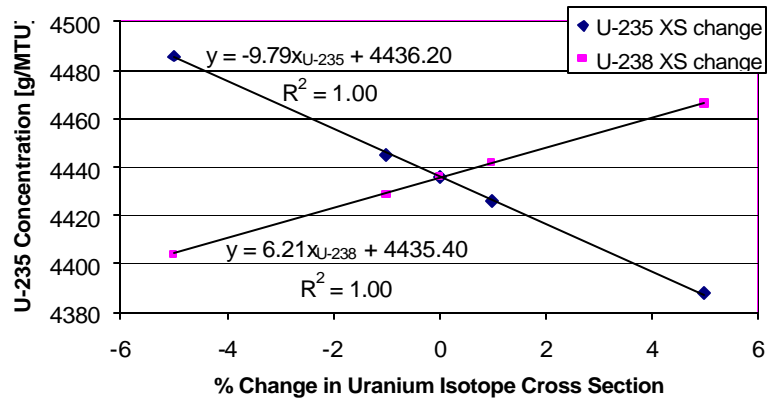
Figure 24. Effect of uranium cross sections on the U-236 concentration at 3093 MWD/MTU
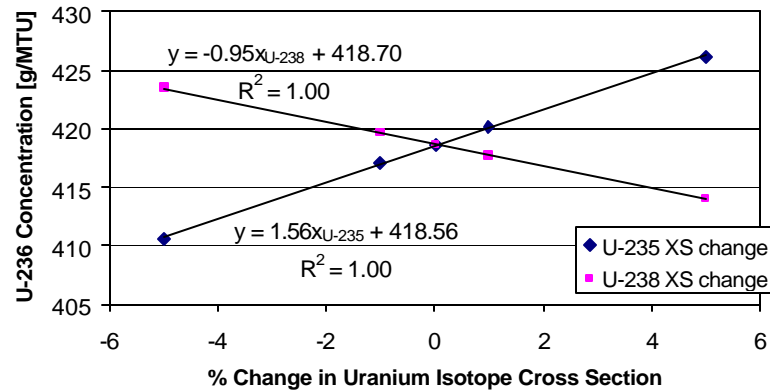
Figure 25. Effect of uranium cross sections on the U-238 concentration at 3093 MWD/MTU
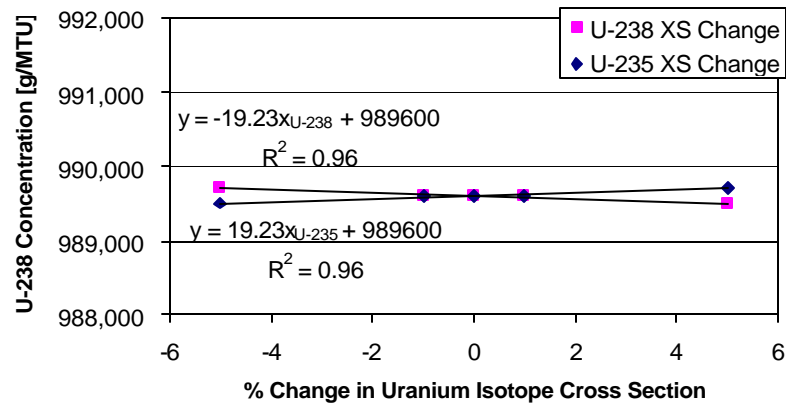
Figure 26-Figure 30 graphically display the effect of changes to the uranium cross sections on the plutonium isotope concentrations at the end of irradiation. All data is for a burnup of 3093 MWD/MTU. Both Pu-238 and Pu-242 had nearly flat lines for changes in the U-238 cross section data. The Pu-238 is insensitive to the U-238 cross section because it is produced from U-235 and not U-238. Pu-241 and Pu-242 are less

sensitive to the U-238 cross section due to the competition for neutron absorption between the isotopes in the system and the small concentrations for these isotopes. Thus, these isotopes at the end of irradiation (EOI) are less sensitive to the uranium cross sections than other isotopes of plutonium. The Pu-239, however, is much more sensitive to changes in the uranium cross sections. Like the uranium isotopic concentrations, the figures below show that there is a linear dependence of the plutonium isotopic concentrations to the values of the U-235 and U-238 cross sections. This observation is again important to the implications of error propagation and quantifying the sensitivity of the results to these parameters.

Figure 26. Effect of uranium cross sections on the Pu-238 concentration at 3093 MWD/MTU
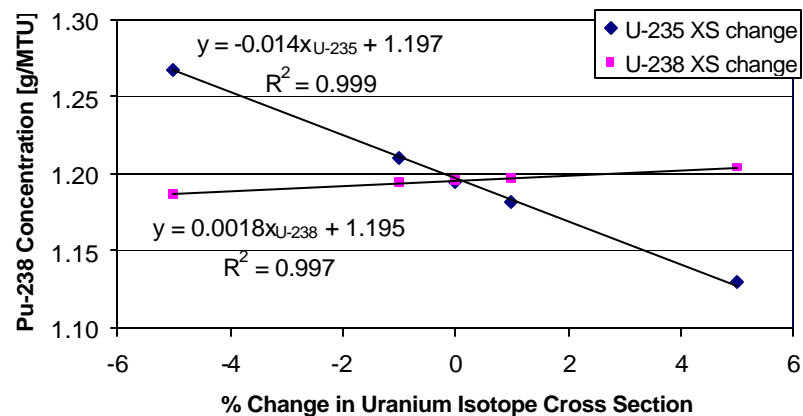
Figure 27. Effect of uranium cross sections on the Pu-239 concentration at 3093 MWD/MTU
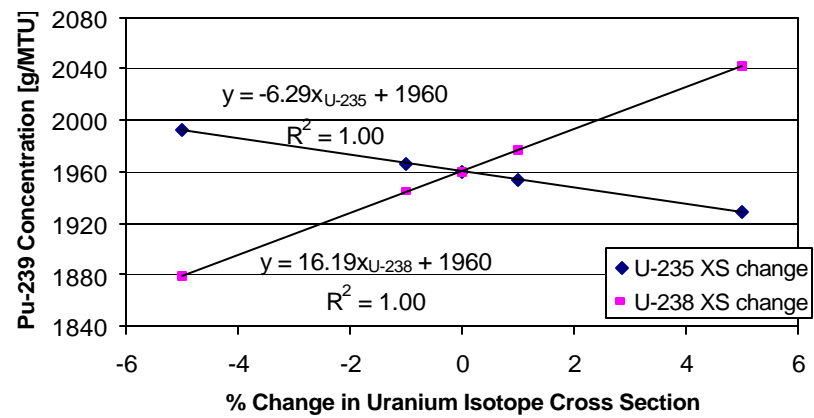
Figure 28. Effect of uranium cross sections on the Pu-240 concentration at 3093 MWD/MTU
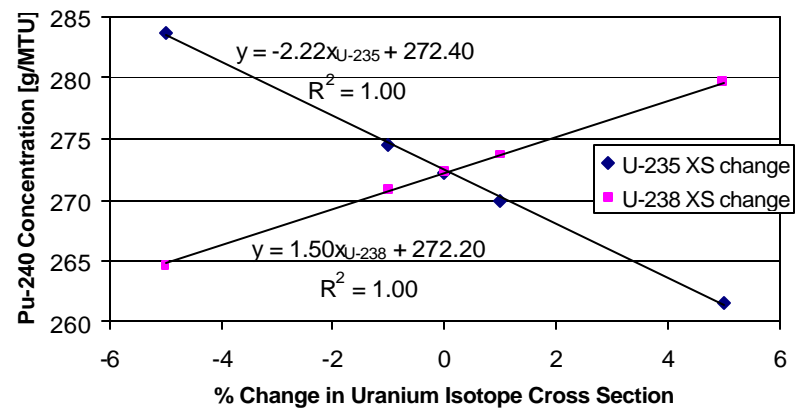
Figure 29. Effect of uranium cross sections on the Pu-241 concentration at 3093 MWD/MTU
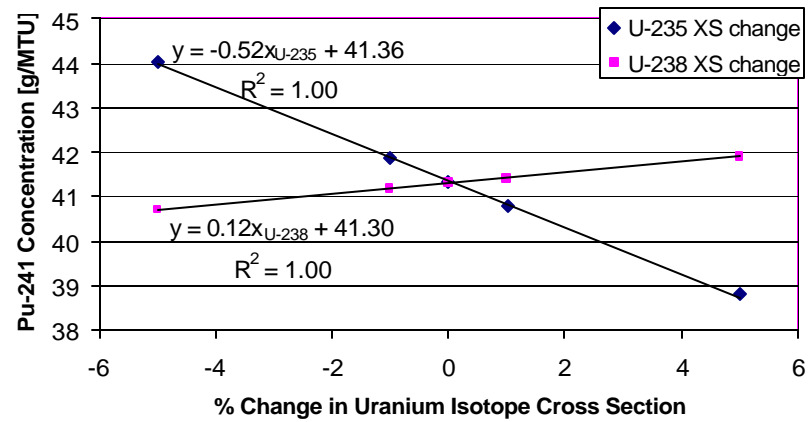
Figure 30. Effect of uranium cross sections on the Pu-242 concentration at 3093 MWD/MTU
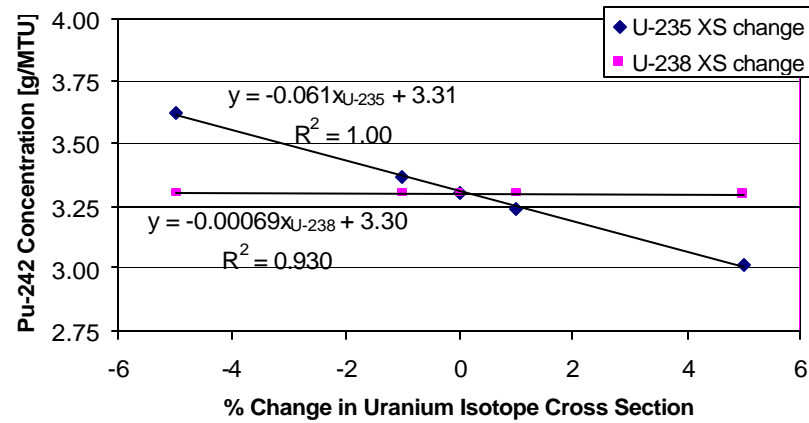
Figure 31. Effect of uranium cross sections on total plutonium concentration at 3093 MWD/MTU
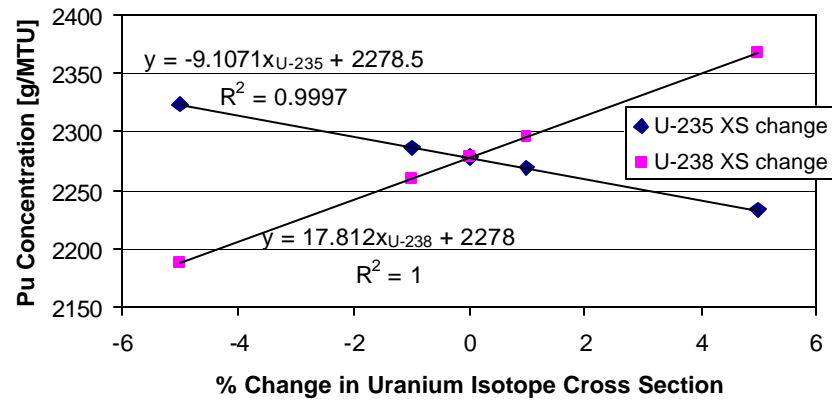
Figure 31 shows the effect of the uranium cross sections on the total plutonium concentration in the fuel rod. The Pu-238 and U-235 concentrations were most sensitive to the U-235 cross section. Pu-239 is most sensitive to the U-238 cross section. The dominant characteristic for these sensitivities is neutron absorption in the uranium isotopes. The change of the absorption cross sections directly affects the plutonium buildup and depletion. This sensitivity is largest for the plutonium isotopes that are closest in mass to the uranium isotopes and produced first in the plutonium buildup chain.

Changes in Trace Elements of the Graphite

The elements suitable for use as indicators for the GIRM method must satisfy several conditions. Most importantly, there must be a measurable quantity of the impurity in the graphite. The activation products of the elements must include other stable isotopes or long-lived isotopes such as those with half-lives on the order of thousands of years. The changes in the concentrations of these trace isotopes as a function of fuel burnup must also be significant. This produces a meaningful correlation between an isotopic ratio and the neutron fluence.

Several different elements were studied in the graphite for the sensitivity analysis including chlorine, titanium, vanadium, lithium, and sulfur. The standard for each impurity was set to be the concentration used previously in the Monteburns simulation for Grade A UK graphite. ORIGEN decks were then run with the concentration varied one element at a time. For each of the five impurities above there were simulations for the standard amount of each element and also twice, one-half, and one-fourth this impurity concentration. Table 12 below lists the quantity of these impurities in grams per kilogram of graphite and the amounts for each variation at beginning-of-life (BOL) where the value of $N(t=0)$ is the concentration found in Grade A UK graphite.

Table 12. Mass in grams of trace elements per kilogram of graphite used in the sensitivity analysis

Element	0.25*N(t=0)	0.5*N(t=0)	N(t=0)	2*N(t=0)
Cl	2.499E-04	4.998E-04	9.997E-04	1.999E-03
Ti	1.999E-03	3.999E-03	7.997E-03	1.599E-02
V	9.997E-03	1.999E-02	3.999E-02	7.998E-02
Li	9.105E-05	1.821E-04	3.642E-04	7.284E-04
S	1.250E-02	2.499E-02	4.998E-02	9.997E-02

Chlorine

To begin looking at input parameters in the graphite, the initial impurity concentration for chlorine was varied to see the effect on the element's isotopic ratios after irradiation. Chlorine has properties desired for use in GIRM. Two stable isotopes— Cl-35 and Cl-37—are found in nature. The isotope Cl-36 is produced following a neutron capture reaction in Cl-35. Cl-36 is a long-lived isotope with a half-life of approximately 301,000 years. The isotopic ratios are also well-correlated to the fuel burnup, or neutron fluence, in the reactor. Figure 32 displays the ratio of the isotopic concentrations for chlorine in the graphite as a function of the fuel. The Cl-36/Cl-37

ratio also has a significant correlation to the neutron fluence in the reactor and its linear behavior would make it applicable over all fluence ranges.

Figure 32. Chlorine isotopic ratios in the graphite moderator as a function of fuel burnup for 9.994E-10 g of chlorine per kg of graphite at BOL
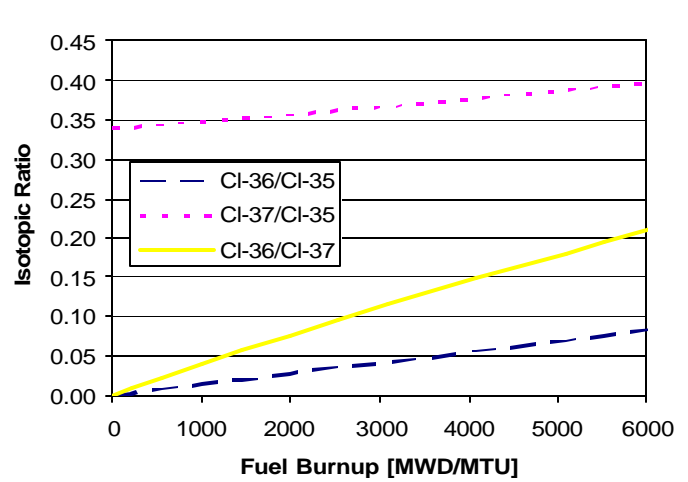
Data was gathered from four ORIGEN simulations of graphite flux irradiations with the various initial concentrations for chlorine as listed above in Table 12. The sensitivity of the measured results of the chlorine isotope ratios after irradiation to the chlorine initial elemental concentrations is shown in Table 13. The percent change was calculated based on (a similar equation was used for each isotopic ratio for each element that will be described later):

$$\%\,Change = \left[\frac{{}^{37}Cl/{}^{35}Cl_{\text{Fraction}Cl} - {}^{37}Cl/{}^{35}Cl_{N(t=0)Cl}}{{}^{37}Cl/{}^{35}Cl_{N(t=0)Cl}} \right] \times 100 \tag{17}$$

where $^{37}Cl/^{35}Cl_{FractionCl}$ is the isotope ratio for a chlorine amount of either 0.25, 0.5 or 2 times N(t=0) at BOL and $^{37}Cl/^{35}Cl_{N(t=0)Cl}$ is the isotope ratio for a chlorine amount of N(t=0) at BOL.

The largest change came from the ratio of chlorine's isotopes, Cl-37/Cl-36, which varied at the greatest by 0.23% at the smallest concentration. However, all isotopic ratios remained essentially the same for chlorine. The Cl-36/Cl-35 ratio only had errors as large as 0.15%, and the Cl-37/Cl-35 ratio varied at most by 0.09%. The trend in the Cl-36/Cl-35 and Cl-37/Cl-35 changes can be seen in Figure 33 where the standard initial chlorine concentration, N(t=0), in the graphite was 9.997E-04 grams per kilogram of graphite.

Table 13. ORIGEN results for chlorine isotopic ratios in the graphite with varying initial chlorine concentrations and a fuel burnup of 3093 MWD/MTU

	N(t=0)	0.25*N(t=0)	% Change	0.5*N(t=0)	% Change	2*N(t=0)	% Change
Cl-36/35	0.0430	0.0429	-0.15%	0.0430	0.00%	0.0430	0.02%
Cl-37/35	0.3523	0.3526	0.08%	0.3526	0.09%	0.3524	0.03%
Cl-37/36	8.197	8.216	0.23%	8.205	0.09%	8.198	0.01%

Figure 33. Chlorine isotopic ratios in the graphite as a function of initial chlorine concentration and a fuel burnup of 3093 MWD/MTU
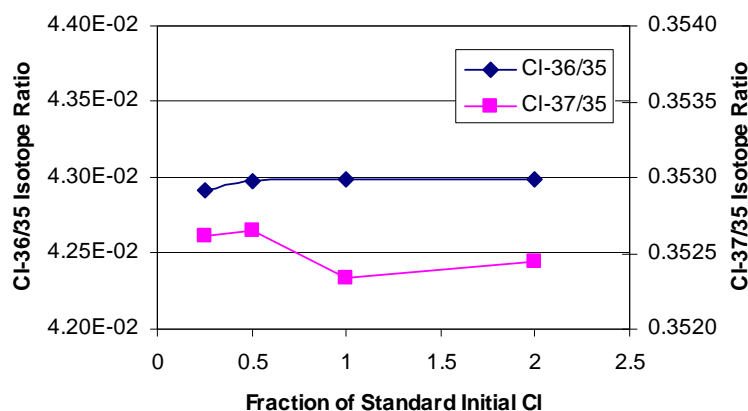
The above results show that the Cl-37/Cl-35 ratio had the least sensitivity of the three ratios to the initial chlorine concentration, but all ratios remained essentially the same for each variation in initial chlorine concentration. Since the initial concentration of the impurities in the graphite is never known, the data suggests that chlorine would make a suitable indicator element in the GIRM method. The ratios produce systematic very small systematic errors of less than one percent for all cases.

Titanium

Titanium has five natural elements: Ti-46, Ti-47, Ti-48, Ti-49, and Ti-50. The Ti-48/Ti-49 ratio was chosen as an indicator ratio in the development of GIRM. For changes in the titanium initial concentration, Table 14 shows the results and percent change of the results as compared to the model with an impurity concentration of N(t=0) for titanium at BOL. The Ti-46/Ti-48 ratio was the least sensitive to change in initial concentration as it varied by less than 1% for any change from the standard initial concentration. The largest change in ratios came after quartering the initial standard concentration of titanium and resulted in changes of approximately 19% and 24%. The ratios of Ti-47/Ti-48, Ti-49/Ti-48, and Ti-50/Ti-48 decreased as the amount of initial titanium increased. The trend of this change can be seen in Figure 34 where the standard initial titanium concentration, N(t=0), was 7.997E-03 grams per kilogram of graphite.

Table 14. ORIGEN results for titanium isotopic ratios in the graphite with varying initial titanium concentrations and a fuel burnup of 3093 MWD/MTU

	N(t=0)	0.25*N(t=0)	% Change	0.5*N(t=0)	% Change	2*N(t=0)	% Change
Ti-46/48	0.108	0.108	0.08%	0.108	0.02%	0.108	0.05%
Ti-47/48	0.099	0.10	0.09%	0.10	0.02%	0.10	0.05%
Ti-49/48	0.083	0.08	0.15%	0.083	0.04%	0.083	0.03%
Ti-50/48	0.074	0.074	0.05%	0.074	0.02%	0.074	0.05%

Figure 34. Titanium isotopic ratios in the graphite as a function of initial titanium concentration and a fuel burnup of 3093 MWD/MTU
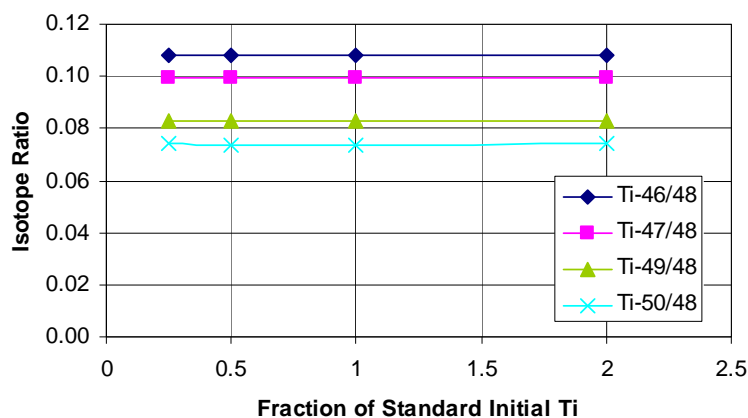
A competing (n,α) reaction produces titanium from chromium. When the concentration of titanium is small enough, this reaction will dominate the titanium production in the graphite rather than the titanium concentration being solely dependent on the neutron absorptions in its own isotopes. Thus, if the concentration of chromium is significantly higher than the titanium in the graphite, the same results and stability of the titanium ratios would not be seen.

The results suggest that the titanium ratios could be used in the GIRM analysis and without large systematic errors as the ratios change little for this particular graphite. However, according to Figure 35, the Ti-46/Ti-48 ratio would not be effective as an

indicator ratio since it does not have a strong correlation to neutron fluence as is seen with the Ti-49/Ti-48 ratio. The Ti-46/Ti-48 ratio remains nearly constant throughout the operation of the reactor and would introduce much larger errors above.

Figure 35. Ti-46/Ti-48 and Ti-49/Ti-48 isotopic ratios in the graphite moderator as a function of fuel burnup
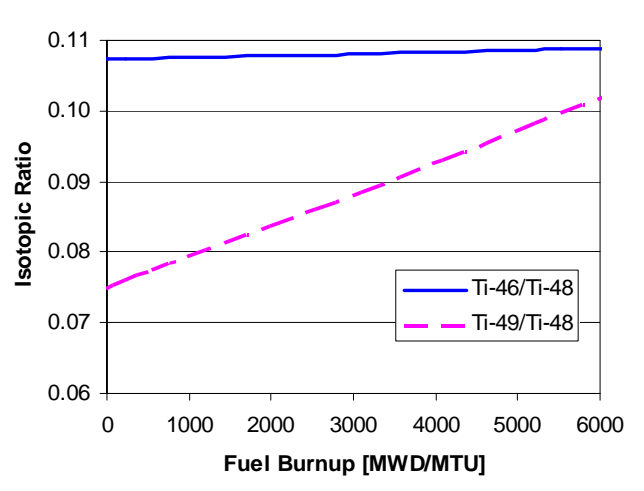
Sulfur

Another isotope considered in the graphite impurity analysis was sulfur. Sulfur has four stable isotopes: S-32, S-33, S-34, and S-36. S-35 which is produced by neutron capture in S-34 has a half-life of 87.2 days making it unsuitable for use with GIRM since it would likely have decayed away prior to sampling and knowledge of the actual shutdown date(s) of the reactor would be important. The initial amount of sulfur in the graphite taken to be the standard, N(t=0), was 4.998E-02 grams per kilogram of graphite. This concentration was altered the same as the cases before: one-fourth, one-half and double the initial standard amount. The ORIGEN simulation was run for each initial concentration with neutron flux irradiations corresponding to a fuel burnup of 3093 MWD/MTU. The results for the isotopic ratios after irradiation are shown in Table 15.

The initial concentration of the sulfur had almost a negligible effect on the isotopic ratios at the end of irradiation.

Table 15. ORIGEN results for sulfur isotopic ratios in the graphite with varying initial sulfur concentrations and a fuel burnup of 3093 MWD/MTU

	N(t=0)	0.25*N(t=0)	% Change	0.5*N(t=0)	% Change	2*N(t=0)	% Change
S-33/32	0.00867	0.00867	0.07%	0.00866	-0.02%	0.00867	0.02%
S-34/32	0.04706	0.04710	0.08%	0.04711	0.09%	0.04708	0.04%
S-34/33	5.430	5.431	0.01%	5.436	0.11%	5.431	0.02%

From these results, it is possible that sulfur could be another useful element for use in the GIRM method. However, the data in Figure 36 shows that the sulfur ratios do not possess a strong correlation to the fuel burnup. The ratios of S-33/S-32 and S-34/S-32 remain essentially constant as the neutron fluence increases. The S-34/S-33 ratio does demonstrate a linear relationship to the fuel burnup; however, the equation shown in the figure for the line demonstrates that the slope of the line is extremely small. Thus, the sulfur ratios could introduce much larger errors in the results if they are used in GIRM.

Figure 36. Sulfur isotopic ratios in the graphite moderator as a function of fuel burnup
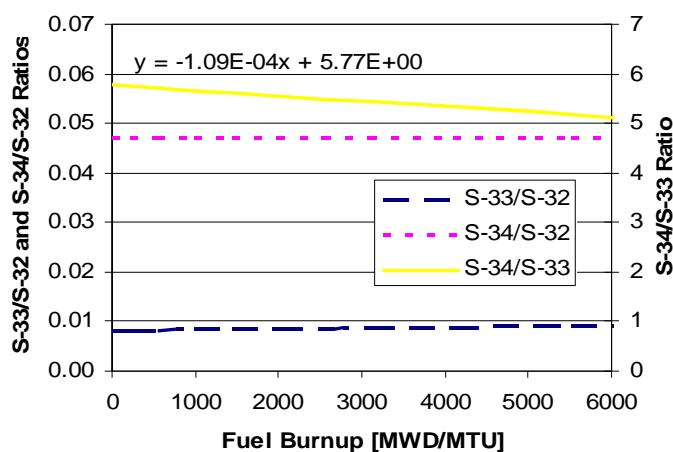
Lithium

Another indicator ratio in the GIRM method is the Li-6/Li-7 isotopic ratio. Both Li-6 and Li-7 are the only two isotopes of lithium found in nature, and all other isotopes of lithium have half-lives of less than one second. Figure 37 displays the ratio value in the graphite moderator as a function of the fuel burnup. This ratio, which follows an exponential curve, is used for low to intermediate fluence ranges since the value of the ratio is the most sensitive to the fluence in this range.

Figure 37. Li-6/Li-7 isotopic ratio in the graphite moderator as a function of fuel burnup
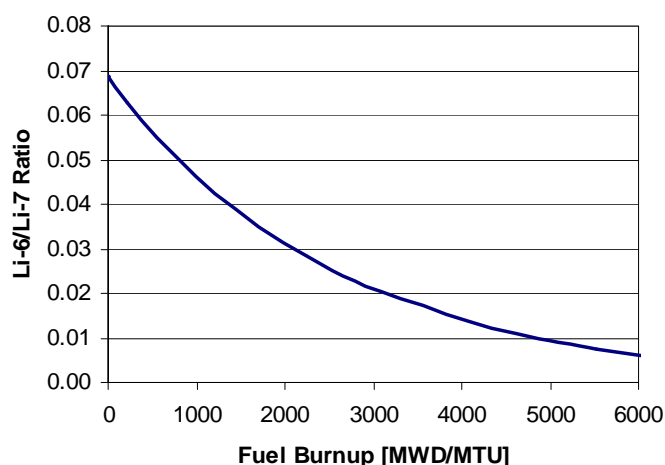
Similar ORIGEN simulations were completed as for the previous elements with one-fourth, one-half and double the initial standard amount of lithium in the graphite. An initial amount of 3.642E-04 grams of lithium per kilogram of graphite was taken to be the standard. Table 16 shows the results of these simulations for a burnup of 3093 MWD/MTU. As compared to the titanium analysis which had almost a 25 percent

change for one isotopic ratio, the Li-6/Li-7 ratio results do not demonstrate a huge variation—less than two percent for all cases. On the other hand, the curve of the data which Figure 38 displays indicates that the relationship of the isotopic ratio to the initial lithium concentration is not linear and could grow increasingly worse with less and less initial lithium concentration. To avoid this, the lithium could be disregarded as an indicator element if its concentration is small enough that it is near the limits of measurability.

Table 16. ORIGEN results for Li-6/Li-7 isotopic ratio in the graphite with varying initial lithium concentrations and a fuel burnup of 3093 MWD/MTU

	$N(t=0)$	$0.25*N(t=0)$	% Change	$0.5*N(t=0)$	% Change	$2*N(t=0)$	% Change
Li-6/Li-7	0.0202	0.0199	-1.74%	0.0201	-0.55%	0.0203	0.25%

Figure 38. Li-6/Li-7 isotopic ratio in the graphite as a function of initial lithium concentration and a fuel burnup of 3093 MWD/MTU
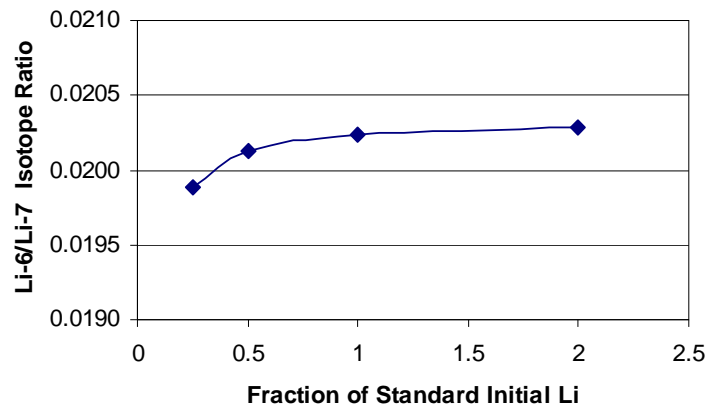
Vanadium

The last isotope chosen for the sensitivity analysis was vanadium. Vanadium is composed of two stable isotopes: V-50 and V-51. The thermal and resonance integral neutron capture cross sections for V-50 are approximately 40 and 60 barns, respectively, which produce V-51. Figure 39 displays the behavior of the V-50/V-51 isotopic ratio with respect to the fuel burnup and indicates that this ratio is highly sensitive to the low fluence range. These characteristics make it a likely candidate as an indicator isotope for GIRM.

Figure 39. V-50/V-51 isotopic ratio in the graphite moderator as a function of fuel burnup
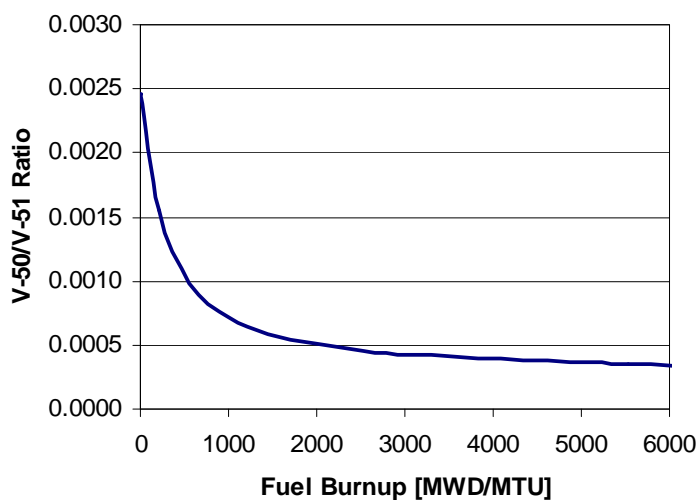
To analyze the sensitivity of the V-50/V-51 ratio to the element's initial concentration, the same procedure was followed as before. The N(t=0) amount of vanadium was 3.999E-02 grams per kilogram of graphite, and ORIGEN simulations were completed with the this amount, one-fourth, one-half and double the initial amount of vanadium in the graphite. Table 17 lists the results from the ORIGEN simulations. The percentage of change for each value is calculated as described above for the chlorine

isotopes. The results show that the V-50/V-51 isotopic ratio is independent to the initial concentration of the natural vanadium in the graphite. Figure 40 graphically displays this data. The large insensitivity of the initial concentration to the isotopic ratio following irradiation makes vanadium useful as an indicator element in the GIRM technique at low fluence ranges where the ratio is strongly correlated to the fuel burnup.

Table 17. ORIGEN results for V-50/V-51 isotopic ratio in the graphite with varying initial vanadium concentrations and a fuel burnup of 3093 MWD/MTU

	N(t=0)	0.25*N(t=0)	% Change	0.5*N(t=0)	% Change	2*N(t=0)	% Change
V-50/V-51	0.002303	0.002302	-0.05%	0.002304	0.01%	0.002303	-0.02%

Figure 40. V-50/V-51 isotopic ratio in the graphite as a function of initial vanadium concentration and a fuel burnup of 3093 MWD/MTU
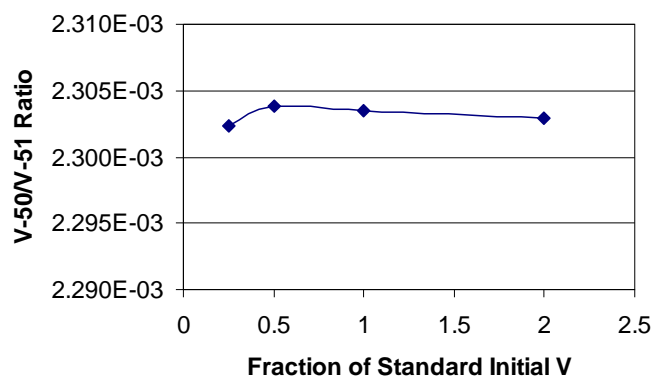
The production of vanadium is also due to the reaction of $^{50}Cr(n,p)^{50}V$ and the beta decay of Ti-51. These reactions create an increase in the V-50 and V-51 concentrations with respect to burnup as shown in Figure 41 when the concentration of chromium is significantly higher than the vanadium. For this scenario where the chromium is significantly higher, the vanadium ratio would have larger sensitivities to

its initial concentration. This graph also compares the vanadium isotopes to Li-6. The aforementioned relationship of the vanadium concentration to other isotopes explains the more extreme change in the V-50/V-51 ratio at low burnups as compared to the Li-6/Li-7 ratio. Li-6 has a much larger cross section for absorption than vanadium, but since the vanadium ratio is a function of isotopes other than its own it changes differently than would be expected if considering only its absorption cross section.

Figure 41. V-50 and V-51 concentrations in the graphite compared to Li-6 as a function of fuel burnup
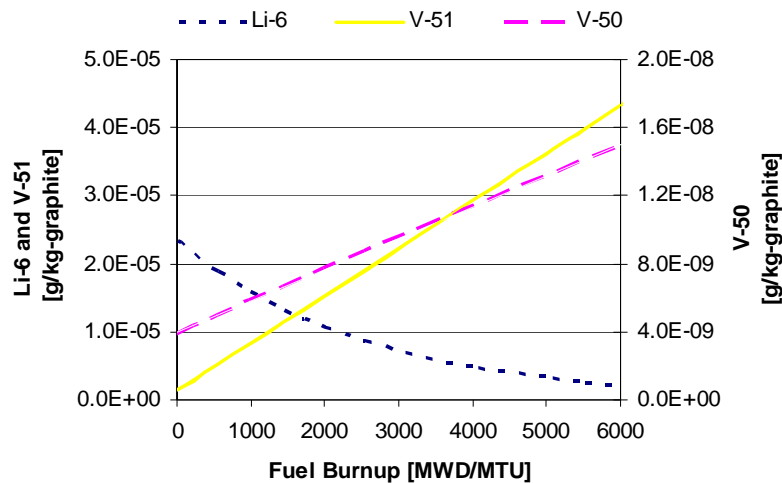
Comparison of Sensitivity Parameters

A sensitivity parameter is useful to compare the expected error from each input parameter to the system results. This sensitivity parameter is defined as:

$$S = \frac{\%\ \text{change in answer}}{\%\ \text{change in parameter}} \tag{18}$$

where S is the sensitivity parameter. The S value provides a basis for comparison between the different input parameters. For input parameters where the answers behaved

linearly (such as the uranium cross section results) this sensitivity parameter is the slope of the lines. In other cases where the results were not linear (such as the results for the titanium ratios), the maximum sensitivity parameter is reported in the following tables. This maximum value assumes a worst-case scenario and is the greatest sensitivity or error seen in the results.

Table 18 displays the sensitivity parameter values associated with each isotope concentration for the aluminum alloy input. These values are also plotted in Figure 42. The isotope with the largest sensitivity to the aluminum alloy concentration in the fuel is the Pu-242. The next most sensitive isotopes are Pu-238 and U-235.

Table 18. Sensitivity parameters for Al alloy input

Isotope	S_{Al}
U-234	-1.129E-02
U-235	-1.454E-02
U-236	-2.869E-03
U-238	-9.953E-03
Pu-238	1.496E-02
Pu-239	-4.090E-03
Pu-240	5.130E-03
Pu-241	1.346E-02
Pu-242	2.394E-02

Figure 42. Sensitivity parameter values for the U and Pu isotopes due to the aluminum alloy input parameter
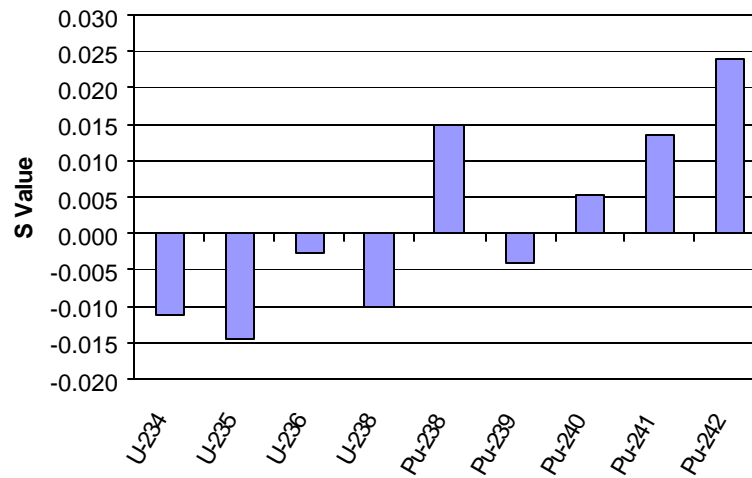
The next set of sensitivity values calculated were for the uranium cross section input parameters. S_{U-235} and S_{U-238} correspond to the sensitivity parameters for the U-235 cross section input and U-238 cross section input, respectively. Figure 43 displays these values graphically. The isotopes most sensitive to the U-235 cross section values were Pu-238, Pu-241, and Pu-242. Pu-239 had the greatest sensitivity to the U-238 cross section.

Table 19. Sensitivity parameters for uranium cross section input

Isotope	S_{U-235}	S_{U-238}
U-234	6.245E-04	2.081E-04
U-235	-2.025E-03	1.580E-03
U-236	3.836E-03	-2.145E-03
U-238	0.000E+00	0.000E+00
Pu-238	-1.240E-02	8.375E-04
Pu-239	-3.052E-03	8.230E-03
Pu-240	-8.015E-03	5.539E-03
Pu-241	-1.290E-02	2.913E-03
Pu-242	-1.873E-02	0.000E+00
Total Pu	-3.856E-03	7.795E-03

Figure 43. Sensitivity parameter values for the U and Pu isotopes due to the U cross section input parameters
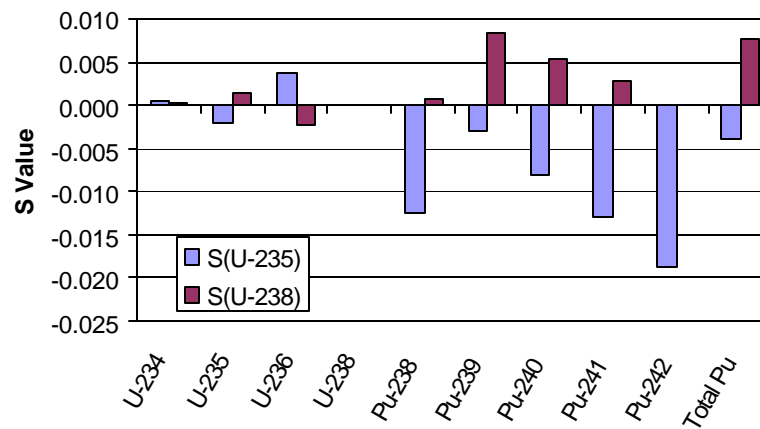
Last, the sensitivity parameters were determined for the various graphite impurity concentration input parameters. These values shown in Table 20 represent the results seen in the above analysis. The largest sensitivities were seen from the concentration of the chlorine, titanium, and vanadium inputs on their respective isotope ratios. Figure 44 displays the results graphically.

Table 20. Sensitivity parameters for the graphite
impurity concentration inputs

Parameter	Ratio	S
Cl Concentration	Cl-36/35	5.647E-03
	Cl-37/35	3.655E-04
	Cl-37/36	-5.274E-03
Ti Concentration	Ti-46/48	-2.628E-03
	Ti-47/48	-2.736E-03
	Ti-49/48	-4.648E-03
	Ti-50/48	-1.365E-03
S Concentration	S-33/32	-3.156E-05
	S-34/32	2.180E-05
	S-34/33	2.229E-02
Li Concentration	Li-6/7	4.826E-02
V Concentration	V-50/51	5.727E-06

Figure 44. Sensitivity parameter values for the different graphite
impurity input parameters
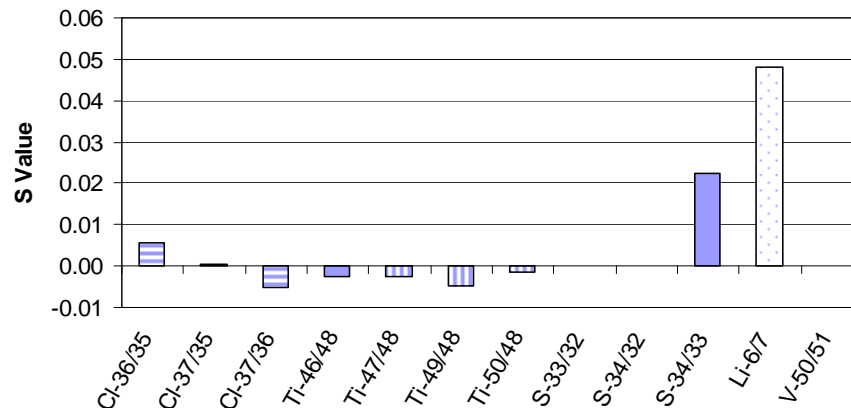
Based on the above sensitivity analysis, the systematic errors associated with
some of the uncertainties in the GIRM method are greater than the errors previously

studied. An earlier error analysis of the reactor physics analysis step in GIRM studied the error associated with the reactor physics calculations and regression model.[14] The conclusion was an error of 1.62% on the plutonium estimation. This error can be minimized by increasing the number of sample taken from the graphite. However, the systematic errors seen from the above parameters cannot be reduced by any such method. For instance, errors due to sensitivities in the graphite impurity concentrations such as in lithium cannot be avoided if the initial impurity concentration is not known. Errors in the system results due to the other input values (cross sections and aluminum alloy concentrations) should be recognized and considered in the estimates from GIRM.

CHAPTER V

SUMMARY AND CONCLUSIONS

A one-group cross section and fission yield library was developed for MAGNOX-style reactors for use with the ORIGEN code. The library was extracted from a Monteburns pin cell model of a Calder Hall reactor fuel pin. The library was successful in calculating uranium and plutonium concentrations for average burnups of 3000 to 4000 MWD/MTU in graphite-moderated reactors. At higher burnups of 5000 and 6000 MWD/MTU, the library resulted in U-235 concentrations within 4% and Pu-239 concentrations within 2% of the Monteburns calculation. Pu-238 and Pu-240 had errors of up to 10% in some cases at these higher burnups.

A sensitivity analysis was also performed using the cross section library to determine the influence of certain parametric changes on the end of irradiation results for an average fuel burnup of 3093 MWD/MTU. Input parameters were chosen for both the fuel and graphite. Fuel parameters included the initial aluminum alloy concentration in the fuel and the U-235 and U-238 cross section values. Graphite parameters included the initial chlorine, titanium, sulfur, lithium, and vanadium concentrations in the graphite before irradiation. This is important for the error associated with methods such as GIRM where there could exist some uncertainty in the concentration of impurities that are initially in the graphite. Analysis of graphite samples in the Trawsfynydd study discovered such heterogeneity for the titanium concentrations.

The results in the fuel were largely influenced by approximately the same percentage of the change in the parameter, whether it is aluminum alloy concentration or change in U-235 or U-238 cross section. The linear relationships between the end of irradiation concentrations for uranium and plutonium with the change in U-235 and U-238 cross sections is important for error analysis when using the ORIGEN library. For cases where the end of irradiation concentrations of uranium and plutonium are to be estimated for a real scenario and the actual U-235 or U-238 cross sections differ from the ORIGEN library values by a few percent, the error of the answers can easily be

determined with standard error propagation. The slope of the equations also quantifies how sensitive the isotope is to changes in the U-235 or U-238 cross section data.

The changes in the graphite parameters had much greater impacts on the results. The titanium ratios were much more sensitive to the initial titanium concentration than the chlorine ratios were to the initial chlorine concentration. Changes in the initial titanium concentration resulted in double or triple the value of the isotopic ratios in some cases. These ratios are important since Ti-48/Ti-49 was previously determined suitable for use in the GIRM technique by method developers. As an indicator ratio it is used to estimate plutonium production. However, the results of the sensitivity analysis show that plutonium estimates using this ratio could have very large errors if the precise concentration of initial titanium is unknown. As was discovered in the Trawsfynydd study, the concentration of titanium varied significantly throughout the graphite. Thus, the Ti-48/Ti-49 ratio should not be used in the method. The only titanium ratio which is insensitive to the initial titanium concentration is Ti-46/Ti-48. However, this ratio is not useful with GIRM since its value has very little dependence on the neutron fluence.

Sulfur, whose isotopic ratios demonstrated almost no sensitivity at all to changes in the initial sulfur concentration, could be a better tool for use with a technique such as GIRM. Of these ratios, only S-34/S-33 demonstrated a functional dependence to the fuel burnup. The linear behavior of this ratio with respect to the fuel burnup also suggests that the ratio of S-34/S-33 is applicable to all ranges of neutron fluence. Thus, the S-34/S-33 ratio could be a useful substitute for the Ti-49/Ti-48 ratio previously used in GIRM if it meets other requirements such as measurability. The effects of the initial concentration of lithium on its end of irradiation isotopic ratios were much smaller than those seen in titanium. The error remained less than 2% for each variation in the lithium concentration found in the graphite before irradiation. On the other hand, the results of vanadium show that it is not effective as an indicator element in GIRM since the vanadium ratio is highly sensitive to its initial concentration in the graphite.

The systematic errors from these input parameters cannot be reduced by increasing the number of graphite samples. Previous studies of GIRM recognized errors

in the regression models and reactor physics analysis calculations.[14] The errors found in this study are in some cases much larger sensitivities than those found in the previous studies. The largest of these sensitivities were seen for some of the initial graphite impurity concentrations. Since the initial concentration of these impurities is not known, those elements with large sensitivities should be avoided as indicators. Sensitivities for other parameters such as uranium cross section values should be considered in the plutonium estimate obtained from GIRM.

The ORIGEN cross section and fission yield library developed in these studies can be applied to any graphite-moderated reactor. The ORIGEN provides a time-efficient method for a sensitivity analysis and burnup and depletion calculations. The ORIGEN code itself is also advantageous in its simplicity and ease in manipulating cross section data and its ability to change power and neutron flux within the system. However, the ORIGEN code has certain disadvantages. It does not use pointwise neutron cross section data to account for energy dependence of the neutrons. The code also does not possess the capability to accommodate changes in geometry, and the user can only change the system power or neutron flux. Analyses involving various sensitivities to geometry require a more complex code such as Monteburns or WIMS which will result in longer computational times. However, these more complex codes can provide a computational analysis of sensitivities that include the dependence of a particular isotopic ratio to the location in the graphite matrix.

The GIRM technique can provide historical plutonium estimates in a reactor. This and similar techniques are extremely important to nuclear nonproliferation and materials accountability. Graphite-moderated reactors have been in use worldwide since the mid-1900s. These reactors are attractive for several reasons. They use natural uranium metal for fuel. Thus, no enrichment facilities are required for fuel production, and the material does not require as many conversion steps as oxide fuels (ore to UF_6 gas to usable oxide). The high U-238 content and low burnup of the fuel is beneficial for production of weapons-grade plutonium. This creates a large proliferation concern, and demands a useful and accurate verification tool for these facilities.

Additional sensitivity analyses could be performed to determine similar relationships for other isotopes and parametric studies. These relationships provide a basis for the type of input data required to obtain accurate answers using such techniques as GIRM and the error than is associated with its results. These material verifications are essential in material protection, control, and accountability and the assurance of global security.

REFERENCES

1.   Fetter, S. 1993. Nuclear Archaeology: Verifying Declarations of Fissile-Material Production, *Science & Global Security*, 3, 237-259.

2.   Fetter S. 1990. Estimating Plutonium Production from Long-lived Radionuclides in Permanent Structural Components of Production Reactor Cores, *Reversing the Arms Race: How to Achieve and Verify Deep Reductions in the Nuclear Arsenals*, Hippel, F. V. and R. Z. Sagdeev, eds., Gordon and Breach, 83–90.

3.   Charlton, W.S., A. Lumley-Woodyear, and K. Budlong-Sylvester. 2002. "Estimating Discharged Plutonium Using Measurements of Structural Material Activation Products," *Transactions of the American Nuclear Society,* 86, 337-339.

4.   Gesh, C. J. 2004. *A Graphite Isotope Ratio Method Primer—A Method for Estimating Plutonium Production in Graphite Moderated Reactors*, PNNL-14568, Pacific Northwest National Laboratory.

5.   Wood, T. W., B. D. Reid, J. L. Smoot, and J. L. Fuller. 2002. Establishing Confident Accounting for Russian Weapons Plutonium, *The Nonproliferation Review*, 9, 2, 126-137.

6.   International Nuclear Safety Center database, http://www.insc.anl.gov/, Argonne National Laboratory (2000).

7.   Rinard, P. 1991. Neutron Interactions with Matter, Chapter 12 of *Passive Nondestructive Assay of Nuclear Materials,* Reilly, D., N. Ensslin, H. Smith, Jr., and S. Kreiner, eds., U.S. Government Printing Office.

8.   Halsall, M. J. 1996. WIMS7, an Overview. *Proceedings of the International Conference on the Physics of Reactors*, *Mito, Japan, September 16-20*, Japan Atomic Energy Research Institute, B-1.

9.   Briesmeister, J. F. 2000. *MCNP™–A General Monte Carlo N–Particle Transport Code*, LA–13709–M, Los Alamos National Laboratory.

10.   Alcouffe, R. E., R. S. Baker, F. W. Brinkley, D. R. Marr, R. D. O'Dell, and W. F. Walters. 1995. *DANTSYS: A Diffusion Accelerated Neutral Particle Transport Code System*, LA-12969-M, Los Alamos National Laboratory.

11.   Little, W. W. 1991. *1DB, A One-Dimensional Diffusion Code for Nuclear Reactor Analysis*, WHC-EP-0435, Westinghouse Company.

12. Reid, B. D., W. C. Morgan, E. F. Love, Jr., D. C. Gerlach, S. L. Peterson, J. V. Livingston, L. R. Greenwood, and J. P. McNeece. 1999. *Graphite Isotope Ratio Method Development Report: Irradiation Test Demonstration of Uranium as a Low Fluence Indicator*, PNNL-13056, Pacific Northwest National Laboratory.

13. Gerlach, D. C., C. J. Gesh, D. E. Hurley, and S. L. Peterson. 2002. *Standard Low-Fluence Graphite Sample Preparation and Measurement Protocol for SIMS/TIMS GIRM Analysis*, Rev. 2 DRAFT, Pacific Northwest National Laboratory.

14. Heasler, P. G., B. D. Reid, C. J. Gesh, D. C. Gerlach, S. L. Peterson, C. K. Bayne, and T. L. Burr. 2003. *Estimation Procedures and Error Analysis for the Graphite Isotopic Ratio Method (GIRM)*, PNNL-14332, Pacific Northwest National Laboratory.

15. Croff, A. G. 1980. *A User's Manual for the ORIGEN2 Computer Code*, ORNL/TM-7175, Oak Ridge National Laboratory.

16. Croff, A. G. 1983. ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials, *Nuclear Technology*, 62, 335-352.

17. Poston, D. I. and H. R. Trellue. 1999. *User's Manual, Version 2.0 for Monteburns, Version 1.0*, LA-UR-99-4999, Los Alamos National Laboratory.

18. MacFarlane, R. E. and D. W. Muir. 1994. *The NJOY Nuclear Data Processing System: Version 91*, LA-12740-M, Los Alamos National Laboratory.

19. International Atomic Energy Agency Power Reactor Information System database from the IAEA WorldAtom webpage, http://www.iaea.org/programmes/a2/, International Atomic Energy Agency (2000).

20. International Atomic Energy Agency. 1965. *The Directory of Nuclear Reactors*, Volume X, IAEA, Vienna, Austria.

21. T-2 Nuclear Information Service, Group T-16 (Nuclear Physics) of the Theoretical Division of the Los Alamos National Laboratory, Index to ENDF/B-VI Neutron Data, http://t2.lanl.gov/cgi-bin/nuclides/endind (01/09/1997).

22. National Nuclear Data Center, Brookhaven National Laboratory, Evaluated Nuclear Data File (ENDF), http://www.nndc.bnl.gov/exfor3/ endf00.htm (02/16/2006).

23. Reid, B. D., D. C. Gerlach, P. G. Heasler, and J. V. Livingston. 1997. *Trawsfynydd Plutonium Estimate*, PNNL Report, Pacific Northwest National Laboratory.

24. Heasler, P. G., T. Burr, B. Reid, C. Gesh, and C. Bayne. 2004. Estimation Procedures and Error Analysis for Inferring the Total Plutonium Produced by a Graphite-Moderated Reactor, *Proceedings of the 4th International Conference on Sensitivity Analysis of Model Output (SAMO 2004)*, Santa Fe, New Mexico, March 8-11, 2004.

APPENDIX A

MONTEBURNS PIN CELL INPUT DECKS

MCNP Input File

```
Calder Hall Fuel Pin
c
c  -------- Cell Cards --------
c
111    1 -18.170 -13  12 -18                 imp:n=1  $Fuel
112    2  -1.740 -13  18 -19                 imp:n=1  $Top Clad Plug
113    2  -1.740 -13  11 -12                 imp:n=1  $Bottom Clad Plug
114    2  -1.740  13 -14  11 -19             imp:n=1  $Clad
115    3  -0.013  14 -15  11 -19             imp:n=1  $Coolant Channel
118    4  -1.700  15  20 -21 22 -23 11 -19   imp:n=1  $Moderator
119    0          -20:21:-22:23:-11:19       imp:n=0  $Universe Around
Pin

c
c  -------- Surface Cards --------
c
*11   pz    0.0000                           $Bottom of Rod
12    pz    2.1167                           $Bottom of Fuel
13    c/z  10.1500   10.1500    1.4605       $Fuel Outer Radius
14    c/z  10.1500   10.1500    1.61925      $Clad Outer Radius
15    c/z  10.1500   10.1500    5.3000       $Graphite Inner Radius
18    pz  103.7167                           $Top of Fuel
*19   pz  105.8334                           $Top of Rod
*20   px    0.0000                           $Left x-Plane for Cell
*21   px   20.3000                           $Right x-Plane for Cell
*22   py    0.0000                           $Bottom y-Plane for Cell
*23   py   20.3000                           $Top y-Plane for Cell

c
c  -------- Data Cards --------
c
mode:n
kcode 3000 1.0 50 500
ksrc  10.1500  10.1500  50.0
c      0.3      0.3     50.0
m1    92234.34c 0.000055
      92235.34c 0.007200
      92238.34c 0.992745          $Natural Uranium Metal
      94238.34c 1e-20
      94239.34c 1e-20
      94240.34c 1e-20
      94241.34c 1e-20
      94242.34c 1e-20
```

```
m2     12000.33c  0.988581
       13027.32c  0.011141
        4009.32c  0.000278              $MAGNOX Clad
m3      6000.32c  0.333333
        8016.31c  0.666667              $Carbon Dioxide
m4      3006.32c  -2.33911E-08
        3007.32c  -3.40810E-07
        4009.32c  -1.00000E-18
        5010.32c  -2.94804E-09
        5011.32c  -1.30470E-08
        6000.32c  -9.99667E-01
       13027.32c  -6.99783E-06
       14028.32c  -7.34756E-05
       14029.32c  -3.85332E-06
       14030.32c  -2.64591E-06
       16000.32c  -4.99845E-05
       17000.32c  -9.99690E-07
       20000.32c  -7.99752E-05
       22000.32c  -7.99752E-06
       23000.32c  -3.99876E-05
       24050.32c  -1.04309E-06
       24052.32c  -2.09183E-05
       24053.32c  -2.41764E-06
       24054.32c  -6.13150E-07
       26054.32c  -1.41095E-06
       26056.32c  -2.29682E-05
       26057.32c  -5.39925E-07
       26058.32c  -7.31134E-08
       28058.32c  -4.03061E-06
       28060.32c  -1.60605E-06
       28061.32c  -7.09857E-08
       28062.32c  -2.29986E-07
       28064.32c  -6.04963E-08
       30000.32c  -3.99876E-07
       56130.32c  -1.00241E-08
       56132.32c  -9.69822E-09
       56134.32c  -2.35603E-07
       56135.32c  -6.47376E-07
       56136.32c  -7.77023E-07
       56137.32c  -1.11920E-06
       56138.32c  -7.19795E-06
       61147.50c  -1.00000E-20
```

Monteburns Input File

```
Calder Hall Fuel Pin, 6.075 GWD/MTU burnup @ 1.38 W/g
PC
2                       !Number of MCNP materials to burn
1                       !MCNP material number #1
4                       !MCNP material number #4
680.8426                !Material #1 volume (cc), input 0 to use mcnp
value
34273.37                !Material #4 volume (cc), input 0 to use mcnp
value
0.01706                 !Power in MWt
-196.0                  !Recov. energy/fis (MeV); if negative use for
U235, ratio other isos
0                       !Total number of days burned (used if no feed
file)
57                      !Number of outer burn steps
10                      !Number of internal burn steps (multiple of 10)
1                       !Number of predictor steps (+1 on first step), 1
usually sufficient
0                       !Step number to restart after (0=beginning)
candunau                !Origen2 library
/packages/origen/origen22/libs  !Location of origen libs
0.005                   !fractional importance (track isos with
abs,fis,atom,mass fraction)
1                       !Intermediate keff calc. 0) No 1) Yes
8                       !Number of automatic tally isotopes, followed by
list.
92234.34c
92235.34c
92238.34c
94238.34c
94239.34c
94240.34c
94241.34c
94242.34c
36                      !Number of automatic tally isotopes, followed by
list.
3006.32c
3007.32c
4009.33c
5010.32c
5011.32c
6000.32c
13027.32c
14028.32c
14029.32c
14030.32c
16000.32c
17000.32c
20000.32c
```

```
22000.32c
23000.32c
24050.32c
24052.32c
24053.32c
24054.32c
26054.32c
26056.32c
26057.32c
26058.32c
28058.32c
28060.32c
28061.32c
28062.32c
28064.32c
30000.32c
56130.32c
56132.32c
56134.32c
56135.32c
56136.32c
56137.32c
56138.32c
```

Monteburns Feed File

```
TimeStep Feed#PowFr mat#BeginAndEnd(-
1=previous)RemovalGroup#FractionF.P.Removed
 (i4) (f8.2), (f7.3) (i4)(i4)  grams/day(2f8.1)(i4)(f7.3)
   1   1  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   2   1  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   3  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   4  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   5  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   6  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   7  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   8  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
   9  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  10  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  11  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  12  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  13  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  14  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  15  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  16  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  17  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  18  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  19  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  20  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  21  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  22  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
  23  80  1.000   1   0   0.0    0.0   0  0.000 0   0.00   0 0.00
               4   0   0.0    0.0   0  0.000 0   0.00   0 0.00
```

24	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
25	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
26	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
27	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
28	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
29	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
30	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
31	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
32	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
33	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
34	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
35	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
36	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
37	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
38	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
39	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
40	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
41	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
42	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
43	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
44	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
45	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
46	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
47	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
48	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00
49	80	1.000	1	0	0.0	0.0	0	0.000	0	0.00	0	0.00
			4	0	0.0	0.0	0	0.000	0	0.00	0	0.00

```
50  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
51  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
52  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
53  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
54  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
55  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
56  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
57  80  1.000  1  0  0.0   0.0   0  0.000 0  0.00  0 0.00
               4  0  0.0   0.0   0  0.000 0  0.00  0 0.00
0                                ! # of feed specs
0                                ! # of removal groups
```

APPENDIX B

ORIGEN DECKS WITH STANDARD GRAPHITE IMPURITIES

Fuel Power Irradiation

```
-1
-1
-1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                       VAR. XSECT
LIB     0    1 2 3      919 920 921    9    50   0  1   0
INP     1  1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF CALDER HALL FUEL
BUP
IRP        1     1.38    1    2    4 2   BURNUP=   1.38 MWD/MTIHM
IRP        2     1.38    2    2    4 0   BURNUP=   2.76 MWD/MTIHM
IRP      102     1.38    2    2    4 0   BURNUP= 140.76 MWD/MTIHM
IRP      202     1.38    2    2    4 0   BURNUP= 278.76 MWD/MTIHM
IRP      302     1.38    2    2    4 0   BURNUP= 416.76 MWD/MTIHM
IRP      402     1.38    2    2    4 0   BURNUP= 554.76 MWD/MTIHM
IRP      502     1.38    2    2    4 0   BURNUP= 692.76 MWD/MTIHM
IRP      602     1.38    2    2    4 0   BURNUP= 830.76 MWD/MTIHM
IRP      702     1.38    2    2    4 0   BURNUP= 968.76 MWD/MTIHM
IRP      802     1.38    2    2    4 0   BURNUP=1106.76 MWD/MTIHM
IRP      902     1.38    2    2    4 0   BURNUP=1244.76 MWD/MTIHM
IRP     1002     1.38    2    2    4 0   BURNUP=1382.76 MWD/MTIHM
IRP     1102     1.38    2    2    4 0   BURNUP=1520.76 MWD/MTIHM
IRP     1202     1.38    2    2    4 0   BURNUP=1658.76 MWD/MTIHM
IRP     1302     1.38    2    2    4 0   BURNUP=1796.76 MWD/MTIHM
IRP     1402     1.38    2    2    4 0   BURNUP=1934.76 MWD/MTIHM
IRP     1502     1.38    2    2    4 0   BURNUP=2072.76 MWD/MTIHM
IRP     1602     1.38    2    2    4 0   BURNUP=2210.76 MWD/MTIHM
IRP     1702     1.38    2    2    4 0   BURNUP=2348.76 MWD/MTIHM
IRP     1802     1.38    2    2    4 0   BURNUP=2486.76 MWD/MTIHM
IRP     1902     1.38    2    2    4 0   BURNUP=2624.76 MWD/MTIHM
IRP     2002     1.38    2    2    4 0   BURNUP=2762.76 MWD/MTIHM
IRP     2102     1.38    2    2    4 0   BURNUP=2900.76 MWD/MTIHM
IRP     2202     1.38    2    2    4 0   BURNUP=3038.76 MWD/MTIHM
IRP     2242     1.38    2    2    4 0   BURNUP=3093.96 MWD/MTIHM
IRP     2302     1.38    2    3    4 0   BURNUP=3176.76 MWD/MTIHM
IRP     2402     1.38    3    3    4 0   BURNUP=3314.76 MWD/MTIHM
IRP     2502     1.38    3    3    4 0   BURNUP=3452.76 MWD/MTIHM
IRP     2602     1.38    3    3    4 0   BURNUP=3590.76 MWD/MTIHM
IRP     2702     1.38    3    3    4 0   BURNUP=3728.76 MWD/MTIHM
IRP     2802     1.38    3    3    4 0   BURNUP=3866.76 MWD/MTIHM
IRP     2902     1.38    3    3    4 0   BURNUP=4004.76 MWD/MTIHM
IRP     2962     1.38    3    3    4 0   BURNUP=4087.56 MWD/MTIHM
IRP     3002     1.38    3    4    4 0   BURNUP=4142.76 MWD/MTIHM
```

```
IRP     3102     1.38    4    4    4 0   BURNUP=4280.76 MWD/MTIHM
IRP     3202     1.38    4    4    4 0   BURNUP=4418.76 MWD/MTIHM
IRP     3302     1.38    4    4    4 0   BURNUP=4556.76 MWD/MTIHM
IRP     3402     1.38    4    4    4 0   BURNUP=4694.76 MWD/MTIHM
IRP     3502     1.38    4    4    4 0   BURNUP=4832.76 MWD/MTIHM
IRP     3602     1.38    4    4    4 0   BURNUP=4970.76 MWD/MTIHM
IRP     3682     1.38    4    4    4 0   BURNUP=5081.16 MWD/MTIHM
IRP     3702     1.38    4    5    4 0   BURNUP=5108.76 MWD/MTIHM
IRP     3802     1.38    5    5    4 0   BURNUP=5246.76 MWD/MTIHM
IRP     3902     1.38    5    5    4 0   BURNUP=5384.76 MWD/MTIHM
IRP     4002     1.38    5    5    4 0   BURNUP=5522.76 MWD/MTIHM
IRP     4102     1.38    5    5    4 0   BURNUP=5660.76 MWD/MTIHM
IRP     4202     1.38    5    5    4 0   BURNUP=5798.76 MWD/MTIHM
IRP     4302     1.38    5    5    4 0   BURNUP=5936.76 MWD/MTIHM
IRP     4402     1.38    5    5    4 0   BURNUP=6074.76 MWD/MTIHM
BUP
RDA
RDA     ***** OUTPUT MODULE *****
TIT      CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT      5    1     -1     0
END
2  922340    54.08   922350   7109.71   922380 992836.22        0     0.0
 0
```

Graphite Neutron Flux Irradiation

```
  -1
  -1
  -1
  RDA *  FLUX IRRADIATION OF 1 KG OF GRADE A UK GRAPHITE IN CALDER HALL
REACTOR
  RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
  RDA        DECAY LIB    XSECT LIB                      VAR. XSECT
  LIB    0    1 2 3     922 923 924    9   50    0  1    0
  INP    1  1  -1  -1  1  1
  TIT    FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
  BUP
  IRF       1     1.67E+13    1    2    4 2
  IRF       2     1.68E+13    2    2    4 0
  IRF      82     1.68E+13    2    2    4 0
  IRF     162     1.68E+13    2    2    4 0
  IRF     242     1.67E+13    2    2    4 0
  IRF     322     1.67E+13    2    2    4 0
  IRF     402     1.67E+13    2    2    4 0
  IRF     482     1.66E+13    2    2    4 0
  IRF     562     1.66E+13    2    2    4 0
  IRF     642     1.66E+13    2    2    4 0
  IRF     722     1.66E+13    2    2    4 0
  IRF     802     1.66E+13    2    2    4 0
  IRF     882     1.66E+13    2    2    4 0
  IRF     962     1.65E+13    2    2    4 0
  IRF    1042     1.66E+13    2    2    4 0
  IRF    1122     1.65E+13    2    2    4 0
  IRF    1202     1.66E+13    2    2    4 0
  IRF    1282     1.66E+13    2    2    4 0
  IRF    1362     1.66E+13    2    2    4 0
  IRF    1442     1.66E+13    2    2    4 0
  IRF    1522     1.66E+13    2    2    4 0
  IRF    1602     1.66E+13    2    2    4 0
  IRF    1682     1.66E+13    2    2    4 0
  IRF    1762     1.66E+13    2    2    4 0
  IRF    1842     1.67E+13    2    2    4 0
  IRF    1922     1.67E+13    2    2    4 0
  IRF    2002     1.67E+13    2    2    4 0
  IRF    2082     1.67E+13    2    2    4 0
  IRF    2162     1.68E+13    2    2    4 0
  IRF    2242     1.68E+13    2    2    4 0
  IRF    2322     1.68E+13    2    2    4 0
  IRF    2402     1.69E+13    2    2    4 0
  IRF    2482     1.69E+13    2    2    4 0
  IRF    2562     1.69E+13    2    2    4 0
  IRF    2642     1.69E+13    2    2    4 0
  IRF    2722     1.70E+13    2    2    4 0
  IRF    2802     1.70E+13    2    2    4 0
  IRF    2882     1.70E+13    2    2    4 0
```

```
IRF     2962      1.70E+13     2    2    4 0
IRF     3042      1.71E+13     2    2    4 0
IRF     3122      1.71E+13     2    2    4 0
IRF     3202      1.71E+13     2    2    4 0
IRF     3282      1.71E+13     2    2    4 0
IRF     3362      1.71E+13     2    2    4 0
IRF     3442      1.72E+13     2    2    4 0
IRF     3522      1.72E+13     2    2    4 0
IRF     3602      1.72E+13     2    2    4 0
IRF     3682      1.73E+13     2    2    4 0
IRF     3762      1.73E+13     2    2    4 0
IRF     3842      1.73E+13     2    2    4 0
IRF     3922      1.73E+13     2    2    4 0
IRF     4002      1.74E+13     2    2    4 0
IRF     4082      1.74E+13     2    2    4 0
IRF     4162      1.74E+13     2    2    4 0
IRF     4242      1.74E+13     2    2    4 0
IRF     4322      1.75E+13     2    2    4 0
IRF     4402      1.75E+13     2    2    4 0
BUP
RDA
RDA     ***** OUTPUT MODULE *****
TIT     CALDER HALL FUEL - STANDARD BURNUP
BAS     1 KG GRADE A UK GRAPHITE
OPTL     8 8 8 8 5 19*8
OPTA     8 8 8 8 5 19*8
OPTF     8 8 8 8 5 19*8
OUT     2    1      -1     0
END
1  30060    2.339E-05     30070   3.408E-04      40090  1.000E-15     50100
2.948E-06
1  50110    1.305E-05     60120   9.874E+02      60130  1.201E+01    130270
6.998E-03
1  140280   7.348E-02    140290   3.853E-03     140300  2.646E-03    160320
2.367E-06
1  160330   1.927E-08    160340   1.114E-07     160360  5.605E-10    170350
7.469E-10
1  170370   2.525E-10    200400   6.182E-06     200420  4.332E-08    200430
9.255E-09
1  200440   1.463E-07    200460   2.933E-10     200480  1.431E-08    220460
5.066E-09
1  220470   4.668E-09    220480   4.723E-08     220490  3.538E-09    220500
3.457E-09
1  230500   3.919E-09    230510   1.595E-06     240500  1.043E-03    240520
2.092E-02
1  240530   2.418E-03    240540   6.132E-04     260540  1.411E-03    260560
2.297E-02
1  260570   5.399E-04    260580   7.311E-05     280580  4.031E-03    280600
1.606E-03
1  280610   7.099E-05    280620   2.300E-04     280640  6.050E-05    300640
7.595E-11
```

```
1  300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
1  561300   1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360   7.770E-04   561370  1.119E-03   561380  7.198E-03        0
0.0
 0
```

APPENDIX C

INPUT ORIGEN DECKS FOR SENSITIVITY ANALYSIS

0.5 w/o Aluminum Alloy in Fuel

```
 -1
 -1
 -1
RDA *  BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA *  0.5 W/O ALUMINUM ALLOY IN FUEL
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA       DECAY LIB    XSECT LIB                    VAR. XSECT
LIB     0    1 2 3     919 920 921    9    50    0  1    0
INP     1  1  -1  -1  1  1
TIT    IRRADIATION OF ONE METRIC TON OF CALDER HALL FUEL
BUP
IRP       1     1.38    1    2    4 2  BURNUP=   1.38 MWD/MTIHM
IRP       2     1.38    2    2    4 0  BURNUP=   2.76 MWD/MTIHM
IRP     102     1.38    2    2    4 0  BURNUP= 140.76 MWD/MTIHM
IRP     202     1.38    2    2    4 0  BURNUP= 278.76 MWD/MTIHM
IRP     302     1.38    2    2    4 0  BURNUP= 416.76 MWD/MTIHM
IRP     402     1.38    2    2    4 0  BURNUP= 554.76 MWD/MTIHM
IRP     502     1.38    2    2    4 0  BURNUP= 692.76 MWD/MTIHM
IRP     602     1.38    2    2    4 0  BURNUP= 830.76 MWD/MTIHM
IRP     702     1.38    2    2    4 0  BURNUP= 968.76 MWD/MTIHM
IRP     802     1.38    2    2    4 0  BURNUP=1106.76 MWD/MTIHM
IRP     902     1.38    2    2    4 0  BURNUP=1244.76 MWD/MTIHM
IRP    1002     1.38    2    2    4 0  BURNUP=1382.76 MWD/MTIHM
IRP    1102     1.38    2    2    4 0  BURNUP=1520.76 MWD/MTIHM
IRP    1202     1.38    2    2    4 0  BURNUP=1658.76 MWD/MTIHM
IRP    1302     1.38    2    2    4 0  BURNUP=1796.76 MWD/MTIHM
IRP    1402     1.38    2    2    4 0  BURNUP=1934.76 MWD/MTIHM
IRP    1502     1.38    2    2    4 0  BURNUP=2072.76 MWD/MTIHM
IRP    1602     1.38    2    2    4 0  BURNUP=2210.76 MWD/MTIHM
IRP    1702     1.38    2    2    4 0  BURNUP=2348.76 MWD/MTIHM
IRP    1802     1.38    2    2    4 0  BURNUP=2486.76 MWD/MTIHM
IRP    1902     1.38    2    2    4 0  BURNUP=2624.76 MWD/MTIHM
IRP    2002     1.38    2    2    4 0  BURNUP=2762.76 MWD/MTIHM
IRP    2102     1.38    2    2    4 0  BURNUP=2900.76 MWD/MTIHM
IRP    2202     1.38    2    2    4 0  BURNUP=3038.76 MWD/MTIHM
IRP    2242     1.38    2    2    4 0  BURNUP=3093.96 MWD/MTIHM
IRP    2302     1.38    2    3    4 0  BURNUP=3176.76 MWD/MTIHM
IRP    2402     1.38    3    3    4 0  BURNUP=3314.76 MWD/MTIHM
IRP    2502     1.38    3    3    4 0  BURNUP=3452.76 MWD/MTIHM
IRP    2602     1.38    3    3    4 0  BURNUP=3590.76 MWD/MTIHM
IRP    2702     1.38    3    3    4 0  BURNUP=3728.76 MWD/MTIHM
IRP    2802     1.38    3    3    4 0  BURNUP=3866.76 MWD/MTIHM
IRP    2902     1.38    3    3    4 0  BURNUP=4004.76 MWD/MTIHM
IRP    2962     1.38    3    3    4 0  BURNUP=4087.56 MWD/MTIHM
```

```
IRP      3002      1.38     3    4     4 0   BURNUP=4142.76 MWD/MTIHM
IRP      3102      1.38     4    4     4 0   BURNUP=4280.76 MWD/MTIHM
IRP      3202      1.38     4    4     4 0   BURNUP=4418.76 MWD/MTIHM
IRP      3302      1.38     4    4     4 0   BURNUP=4556.76 MWD/MTIHM
IRP      3402      1.38     4    4     4 0   BURNUP=4694.76 MWD/MTIHM
IRP      3502      1.38     4    4     4 0   BURNUP=4832.76 MWD/MTIHM
IRP      3602      1.38     4    4     4 0   BURNUP=4970.76 MWD/MTIHM
IRP      3682      1.38     4    4     4 0   BURNUP=5081.16 MWD/MTIHM
IRP      3702      1.38     4    5     4 0   BURNUP=5108.76 MWD/MTIHM
IRP      3802      1.38     5    5     4 0   BURNUP=5246.76 MWD/MTIHM
IRP      3902      1.38     5    5     4 0   BURNUP=5384.76 MWD/MTIHM
IRP      4002      1.38     5    5     4 0   BURNUP=5522.76 MWD/MTIHM
IRP      4102      1.38     5    5     4 0   BURNUP=5660.76 MWD/MTIHM
IRP      4202      1.38     5    5     4 0   BURNUP=5798.76 MWD/MTIHM
IRP      4302      1.38     5    5     4 0   BURNUP=5936.76 MWD/MTIHM
IRP      4402      1.38     5    5     4 0   BURNUP=6074.76 MWD/MTIHM
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT      CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT      5    1     -1    0
END
2  922340    53.81   922350   7074.16   922380 987872.03        0     0.0
1  130270  5000.00        0      0.0         0      0.0          0     0.0
 0
```

1.0 w/o Aluminum Alloy in Fuel

```
 -1
 -1
 -1
RDA *  BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA *  1.0 W/O ALUMINUM ALLOY IN FUEL
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     919 920 921    9    50    0   1    0
INP     1  1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF CALDER HALL FUEL
BUP
IRP        1    1.38    1    2    4 2   BURNUP=   1.38 MWD/MTIHM
IRP        2    1.38    2    2    4 0   BURNUP=   2.76 MWD/MTIHM
IRP      102    1.38    2    2    4 0   BURNUP= 140.76 MWD/MTIHM
IRP      202    1.38    2    2    4 0   BURNUP= 278.76 MWD/MTIHM
IRP      302    1.38    2    2    4 0   BURNUP= 416.76 MWD/MTIHM
IRP      402    1.38    2    2    4 0   BURNUP= 554.76 MWD/MTIHM
IRP      502    1.38    2    2    4 0   BURNUP= 692.76 MWD/MTIHM
IRP      602    1.38    2    2    4 0   BURNUP= 830.76 MWD/MTIHM
IRP      702    1.38    2    2    4 0   BURNUP= 968.76 MWD/MTIHM
IRP      802    1.38    2    2    4 0   BURNUP=1106.76 MWD/MTIHM
IRP      902    1.38    2    2    4 0   BURNUP=1244.76 MWD/MTIHM
IRP     1002    1.38    2    2    4 0   BURNUP=1382.76 MWD/MTIHM
IRP     1102    1.38    2    2    4 0   BURNUP=1520.76 MWD/MTIHM
IRP     1202    1.38    2    2    4 0   BURNUP=1658.76 MWD/MTIHM
IRP     1302    1.38    2    2    4 0   BURNUP=1796.76 MWD/MTIHM
IRP     1402    1.38    2    2    4 0   BURNUP=1934.76 MWD/MTIHM
IRP     1502    1.38    2    2    4 0   BURNUP=2072.76 MWD/MTIHM
IRP     1602    1.38    2    2    4 0   BURNUP=2210.76 MWD/MTIHM
IRP     1702    1.38    2    2    4 0   BURNUP=2348.76 MWD/MTIHM
IRP     1802    1.38    2    2    4 0   BURNUP=2486.76 MWD/MTIHM
IRP     1902    1.38    2    2    4 0   BURNUP=2624.76 MWD/MTIHM
IRP     2002    1.38    2    2    4 0   BURNUP=2762.76 MWD/MTIHM
IRP     2102    1.38    2    2    4 0   BURNUP=2900.76 MWD/MTIHM
IRP     2202    1.38    2    2    4 0   BURNUP=3038.76 MWD/MTIHM
IRP     2242    1.38    2    2    4 0   BURNUP=3093.96 MWD/MTIHM
IRP     2302    1.38    2    3    4 0   BURNUP=3176.76 MWD/MTIHM
IRP     2402    1.38    3    3    4 0   BURNUP=3314.76 MWD/MTIHM
IRP     2502    1.38    3    3    4 0   BURNUP=3452.76 MWD/MTIHM
IRP     2602    1.38    3    3    4 0   BURNUP=3590.76 MWD/MTIHM
IRP     2702    1.38    3    3    4 0   BURNUP=3728.76 MWD/MTIHM
IRP     2802    1.38    3    3    4 0   BURNUP=3866.76 MWD/MTIHM
IRP     2902    1.38    3    3    4 0   BURNUP=4004.76 MWD/MTIHM
IRP     2962    1.38    3    3    4 0   BURNUP=4087.56 MWD/MTIHM
IRP     3002    1.38    3    4    4 0   BURNUP=4142.76 MWD/MTIHM
IRP     3102    1.38    4    4    4 0   BURNUP=4280.76 MWD/MTIHM
IRP     3202    1.38    4    4    4 0   BURNUP=4418.76 MWD/MTIHM
IRP     3302    1.38    4    4    4 0   BURNUP=4556.76 MWD/MTIHM
IRP     3402    1.38    4    4    4 0   BURNUP=4694.76 MWD/MTIHM
IRP     3502    1.38    4    4    4 0   BURNUP=4832.76 MWD/MTIHM
```

```
IRP       3602      1.38     4    4    4 0   BURNUP=4970.76 MWD/MTIHM
IRP       3682      1.38     4    4    4 0   BURNUP=5081.16 MWD/MTIHM
IRP       3702      1.38     4    5    4 0   BURNUP=5108.76 MWD/MTIHM
IRP       3802      1.38     5    5    4 0   BURNUP=5246.76 MWD/MTIHM
IRP       3902      1.38     5    5    4 0   BURNUP=5384.76 MWD/MTIHM
IRP       4002      1.38     5    5    4 0   BURNUP=5522.76 MWD/MTIHM
IRP       4102      1.38     5    5    4 0   BURNUP=5660.76 MWD/MTIHM
IRP       4202      1.38     5    5    4 0   BURNUP=5798.76 MWD/MTIHM
IRP       4302      1.38     5    5    4 0   BURNUP=5936.76 MWD/MTIHM
IRP       4402      1.38     5    5    4 0   BURNUP=6074.76 MWD/MTIHM
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT       CALDER HALL FUEL - STANDARD BURNUP
BAS       1 MTIHM NAT U
OPTL       8 8 8 8 5 19*8
OPTA       8 8 8 8 5 19*8
OPTF       8 8 8 8 5 19*8
OUT        5    1     -1     0
END
2  922340      53.54   922350   7038.61   922380 982907.85         0      0.0
1  130270 10000.00        0      0.0         0      0.0             0      0.0
 0
```

2.0 w/o Aluminum Alloy in Fuel

```
 -1
 -1
 -1
RDA *  BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA *  2.0 W/O ALUMINUM ALLOY IN FUEL
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA       DECAY LIB    XSECT LIB                        VAR. XSECT
LIB      0    1 2 3     919 920 921     9   50   0  1   0
INP      1  1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF CALDER HALL FUEL
BUP
IRP        1     1.38    1   2   4 2   BURNUP=    1.38 MWD/MTIHM
IRP        2     1.38    2   2   4 0   BURNUP=    2.76 MWD/MTIHM
IRP      102     1.38    2   2   4 0   BURNUP=  140.76 MWD/MTIHM
IRP      202     1.38    2   2   4 0   BURNUP=  278.76 MWD/MTIHM
IRP      302     1.38    2   2   4 0   BURNUP=  416.76 MWD/MTIHM
IRP      402     1.38    2   2   4 0   BURNUP=  554.76 MWD/MTIHM
IRP      502     1.38    2   2   4 0   BURNUP=  692.76 MWD/MTIHM
IRP      602     1.38    2   2   4 0   BURNUP=  830.76 MWD/MTIHM
IRP      702     1.38    2   2   4 0   BURNUP=  968.76 MWD/MTIHM
IRP      802     1.38    2   2   4 0   BURNUP=1106.76 MWD/MTIHM
IRP      902     1.38    2   2   4 0   BURNUP=1244.76 MWD/MTIHM
IRP     1002     1.38    2   2   4 0   BURNUP=1382.76 MWD/MTIHM
IRP     1102     1.38    2   2   4 0   BURNUP=1520.76 MWD/MTIHM
IRP     1202     1.38    2   2   4 0   BURNUP=1658.76 MWD/MTIHM
IRP     1302     1.38    2   2   4 0   BURNUP=1796.76 MWD/MTIHM
IRP     1402     1.38    2   2   4 0   BURNUP=1934.76 MWD/MTIHM
IRP     1502     1.38    2   2   4 0   BURNUP=2072.76 MWD/MTIHM
IRP     1602     1.38    2   2   4 0   BURNUP=2210.76 MWD/MTIHM
IRP     1702     1.38    2   2   4 0   BURNUP=2348.76 MWD/MTIHM
IRP     1802     1.38    2   2   4 0   BURNUP=2486.76 MWD/MTIHM
IRP     1902     1.38    2   2   4 0   BURNUP=2624.76 MWD/MTIHM
IRP     2002     1.38    2   2   4 0   BURNUP=2762.76 MWD/MTIHM
IRP     2102     1.38    2   2   4 0   BURNUP=2900.76 MWD/MTIHM
IRP     2202     1.38    2   2   4 0   BURNUP=3038.76 MWD/MTIHM
IRP     2242     1.38    2   2   4 0   BURNUP=3093.96 MWD/MTIHM
IRP     2302     1.38    2   3   4 0   BURNUP=3176.76 MWD/MTIHM
IRP     2402     1.38    3   3   4 0   BURNUP=3314.76 MWD/MTIHM
IRP     2502     1.38    3   3   4 0   BURNUP=3452.76 MWD/MTIHM
IRP     2602     1.38    3   3   4 0   BURNUP=3590.76 MWD/MTIHM
IRP     2702     1.38    3   3   4 0   BURNUP=3728.76 MWD/MTIHM
IRP     2802     1.38    3   3   4 0   BURNUP=3866.76 MWD/MTIHM
IRP     2902     1.38    3   3   4 0   BURNUP=4004.76 MWD/MTIHM
IRP     2962     1.38    3   3   4 0   BURNUP=4087.56 MWD/MTIHM
IRP     3002     1.38    3   4   4 0   BURNUP=4142.76 MWD/MTIHM
IRP     3102     1.38    4   4   4 0   BURNUP=4280.76 MWD/MTIHM
IRP     3202     1.38    4   4   4 0   BURNUP=4418.76 MWD/MTIHM
IRP     3302     1.38    4   4   4 0   BURNUP=4556.76 MWD/MTIHM
IRP     3402     1.38    4   4   4 0   BURNUP=4694.76 MWD/MTIHM
IRP     3502     1.38    4   4   4 0   BURNUP=4832.76 MWD/MTIHM
```

```
IRP        3602      1.38     4    4     4 0   BURNUP=4970.76 MWD/MTIHM
IRP        3682      1.38     4    4     4 0   BURNUP=5081.16 MWD/MTIHM
IRP        3702      1.38     4    5     4 0   BURNUP=5108.76 MWD/MTIHM
IRP        3802      1.38     5    5     4 0   BURNUP=5246.76 MWD/MTIHM
IRP        3902      1.38     5    5     4 0   BURNUP=5384.76 MWD/MTIHM
IRP        4002      1.38     5    5     4 0   BURNUP=5522.76 MWD/MTIHM
IRP        4102      1.38     5    5     4 0   BURNUP=5660.76 MWD/MTIHM
IRP        4202      1.38     5    5     4 0   BURNUP=5798.76 MWD/MTIHM
IRP        4302      1.38     5    5     4 0   BURNUP=5936.76 MWD/MTIHM
IRP        4402      1.38     5    5     4 0   BURNUP=6074.76 MWD/MTIHM
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT       CALDER HALL FUEL - STANDARD BURNUP
BAS       1 MTIHM NAT U
OPTL       8 8 8 8 5 19*8
OPTA       8 8 8 8 5 19*8
OPTF       8 8 8 8 5 19*8
OUT        5    1     -1    0
END
2  922340     53.00   922350  6967.51   922380 972979.49        0     0.0
1  130270  20000.00       0      0.0        0       0.0          0     0.0
 0
```

One-Fourth Initial Chlorine Concentration

```
 -1
 -1
 -1
RDA  *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA  **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB   XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     922 923 924    9   50   0  1   0
INP     1  1  -1  -1   1  1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1     1.67E+13    1    2    4 2
IRF        2     1.68E+13    2    2    4 0
IRF       82     1.68E+13    2    2    4 0
IRF      162     1.68E+13    2    2    4 0
IRF      242     1.67E+13    2    2    4 0
IRF      322     1.67E+13    2    2    4 0
IRF      402     1.67E+13    2    2    4 0
IRF      482     1.66E+13    2    2    4 0
IRF      562     1.66E+13    2    2    4 0
IRF      642     1.66E+13    2    2    4 0
IRF      722     1.66E+13    2    2    4 0
IRF      802     1.66E+13    2    2    4 0
IRF      882     1.66E+13    2    2    4 0
IRF      962     1.65E+13    2    2    4 0
IRF     1042     1.66E+13    2    2    4 0
IRF     1122     1.65E+13    2    2    4 0
IRF     1202     1.66E+13    2    2    4 0
IRF     1282     1.66E+13    2    2    4 0
IRF     1362     1.66E+13    2    2    4 0
IRF     1442     1.66E+13    2    2    4 0
IRF     1522     1.66E+13    2    2    4 0
IRF     1602     1.66E+13    2    2    4 0
IRF     1682     1.66E+13    2    2    4 0
IRF     1762     1.66E+13    2    2    4 0
IRF     1842     1.67E+13    2    2    4 0
IRF     1922     1.67E+13    2    2    4 0
IRF     2002     1.67E+13    2    2    4 0
IRF     2082     1.67E+13    2    2    4 0
IRF     2162     1.68E+13    2    2    4 0
IRF     2242     1.68E+13    2    2    4 0
IRF     2322     1.68E+13    2    3    4 0
IRF     2402     1.69E+13    3    3    4 0
IRF     2482     1.69E+13    3    3    4 0
IRF     2562     1.69E+13    3    3    4 0
IRF     2642     1.69E+13    3    3    4 0
IRF     2722     1.70E+13    3    3    4 0
IRF     2802     1.70E+13    3    3    4 0
IRF     2882     1.70E+13    3    3    4 0
IRF     2962     1.70E+13    3    3    4 0
IRF     3042     1.71E+13    3    4    4 0
```

```
IRF       3122      1.71E+13     4    4     4 0
IRF       3202      1.71E+13     4    4     4 0
IRF       3282      1.71E+13     4    4     4 0
IRF       3362      1.71E+13     4    4     4 0
IRF       3442      1.72E+13     4    4     4 0
IRF       3522      1.72E+13     4    4     4 0
IRF       3602      1.72E+13     4    4     4 0
IRF       3682      1.73E+13     4    4     4 0
IRF       3762      1.73E+13     4    5     4 0
IRF       3842      1.73E+13     5    5     4 0
IRF       3922      1.73E+13     5    5     4 0
IRF       4002      1.74E+13     5    5     4 0
IRF       4082      1.74E+13     5    5     4 0
IRF       4162      1.74E+13     5    5     4 0
IRF       4242      1.74E+13     5    5     4 0
IRF       4322      1.75E+13     5    5     4 0
IRF       4402      1.75E+13     5    5     4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT       CALDER HALL FUEL - STANDARD BURNUP
BAS       1 MTIHM NAT U
OPTL       8 8 8 8 5 19*8
OPTA       8 8 8 8 5 19*8
OPTF       8 8 8 8 5 19*8
OUT        5    1     -1     0
END
1  30060    2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120  9.874E+02    60130  1.201E+01   130270
6.998E-03
1  140280   7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
2.367E-06
1  160330   1.927E-08   160340  1.114E-07   160360  5.605E-10   170350
1.8673E-10
1  170370 6.3125E-11   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1  200440   1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
5.066E-09
1  220470   4.668E-09   220480  4.723E-08   220490  3.538E-09   220500
3.457E-09
1  230500   3.919E-09   230510  1.595E-06   240500  1.043E-03   240520
2.092E-02
1  240530   2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1  260570   5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1  280610   7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1  300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1  561300   1.002E-05    561320   9.698E-06    561340   2.356E-04    561350
6.474E-04
1  561360   7.770E-04    561370   1.119E-03    561380   7.198E-03          0
0.0
 0
```

One-Half Initial Chlorine Concentration

```
  -1
  -1
  -1
 RDA  *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
 RDA  **   CROSS SECTION LIBRARY = STEP 30 TAPE 9
 RDA        DECAY LIB    XSECT LIB                          VAR. XSECT
 LIB     0    1 2 3     922 923 924    9   50   0  1   0
 INP     1  1  -1  -1   1  1
 TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
 BUP
 IRF        1     1.67E+13    1   2    4 2
 IRF        2     1.68E+13    2   2    4 0
 IRF       82     1.68E+13    2   2    4 0
 IRF      162     1.68E+13    2   2    4 0
 IRF      242     1.67E+13    2   2    4 0
 IRF      322     1.67E+13    2   2    4 0
 IRF      402     1.67E+13    2   2    4 0
 IRF      482     1.66E+13    2   2    4 0
 IRF      562     1.66E+13    2   2    4 0
 IRF      642     1.66E+13    2   2    4 0
 IRF      722     1.66E+13    2   2    4 0
 IRF      802     1.66E+13    2   2    4 0
 IRF      882     1.66E+13    2   2    4 0
 IRF      962     1.65E+13    2   2    4 0
 IRF     1042     1.66E+13    2   2    4 0
 IRF     1122     1.65E+13    2   2    4 0
 IRF     1202     1.66E+13    2   2    4 0
 IRF     1282     1.66E+13    2   2    4 0
 IRF     1362     1.66E+13    2   2    4 0
 IRF     1442     1.66E+13    2   2    4 0
 IRF     1522     1.66E+13    2   2    4 0
 IRF     1602     1.66E+13    2   2    4 0
 IRF     1682     1.66E+13    2   2    4 0
 IRF     1762     1.66E+13    2   2    4 0
 IRF     1842     1.67E+13    2   2    4 0
 IRF     1922     1.67E+13    2   2    4 0
 IRF     2002     1.67E+13    2   2    4 0
 IRF     2082     1.67E+13    2   2    4 0
 IRF     2162     1.68E+13    2   2    4 0
 IRF     2242     1.68E+13    2   2    4 0
 IRF     2322     1.68E+13    2   3    4 0
 IRF     2402     1.69E+13    3   3    4 0
 IRF     2482     1.69E+13    3   3    4 0
 IRF     2562     1.69E+13    3   3    4 0
 IRF     2642     1.69E+13    3   3    4 0
 IRF     2722     1.70E+13    3   3    4 0
 IRF     2802     1.70E+13    3   3    4 0
 IRF     2882     1.70E+13    3   3    4 0
 IRF     2962     1.70E+13    3   3    4 0
 IRF     3042     1.71E+13    3   4    4 0
```

```
IRF      3122      1.71E+13     4    4    4 0
IRF      3202      1.71E+13     4    4    4 0
IRF      3282      1.71E+13     4    4    4 0
IRF      3362      1.71E+13     4    4    4 0
IRF      3442      1.72E+13     4    4    4 0
IRF      3522      1.72E+13     4    4    4 0
IRF      3602      1.72E+13     4    4    4 0
IRF      3682      1.73E+13     4    4    4 0
IRF      3762      1.73E+13     4    5    4 0
IRF      3842      1.73E+13     5    5    4 0
IRF      3922      1.73E+13     5    5    4 0
IRF      4002      1.74E+13     5    5    4 0
IRF      4082      1.74E+13     5    5    4 0
IRF      4162      1.74E+13     5    5    4 0
IRF      4242      1.74E+13     5    5    4 0
IRF      4322      1.75E+13     5    5    4 0
IRF      4402      1.75E+13     5    5    4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT       CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT       5    1      -1    0
END
1  30060    2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120  9.874E+02    60130  1.201E+01    130270
6.998E-03
1  140280   7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
2.367E-06
1  160330   1.927E-08   160340  1.114E-07   160360  5.605E-10   170350
3.7345E-10
1  170370 1.2625E-10   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1  200440   1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
5.066E-09
1  220470   4.668E-09   220480  4.723E-08   220490  3.538E-09   220500
3.457E-09
1  230500   3.919E-09   230510  1.595E-06   240500  1.043E-03   240520
2.092E-02
1  240530   2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1  260570   5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1  280610   7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1  300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03         0
0.0
 0
```

Double Initial Chlorine Concentration

```
 -1
 -1
 -1
RDA  *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA  **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                         VAR. XSECT
LIB    0    1 2 3      922 923 924    9    50    0  1    0
INP    1  1  -1  -1   1   1
TIT    FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1      1.67E+13    1    2    4 2
IRF        2      1.68E+13    2    2    4 0
IRF       82      1.68E+13    2    2    4 0
IRF      162      1.68E+13    2    2    4 0
IRF      242      1.67E+13    2    2    4 0
IRF      322      1.67E+13    2    2    4 0
IRF      402      1.67E+13    2    2    4 0
IRF      482      1.66E+13    2    2    4 0
IRF      562      1.66E+13    2    2    4 0
IRF      642      1.66E+13    2    2    4 0
IRF      722      1.66E+13    2    2    4 0
IRF      802      1.66E+13    2    2    4 0
IRF      882      1.66E+13    2    2    4 0
IRF      962      1.65E+13    2    2    4 0
IRF     1042      1.66E+13    2    2    4 0
IRF     1122      1.65E+13    2    2    4 0
IRF     1202      1.66E+13    2    2    4 0
IRF     1282      1.66E+13    2    2    4 0
IRF     1362      1.66E+13    2    2    4 0
IRF     1442      1.66E+13    2    2    4 0
IRF     1522      1.66E+13    2    2    4 0
IRF     1602      1.66E+13    2    2    4 0
IRF     1682      1.66E+13    2    2    4 0
IRF     1762      1.66E+13    2    2    4 0
IRF     1842      1.67E+13    2    2    4 0
IRF     1922      1.67E+13    2    2    4 0
IRF     2002      1.67E+13    2    2    4 0
IRF     2082      1.67E+13    2    2    4 0
IRF     2162      1.68E+13    2    2    4 0
IRF     2242      1.68E+13    2    2    4 0
IRF     2322      1.68E+13    2    3    4 0
IRF     2402      1.69E+13    3    3    4 0
IRF     2482      1.69E+13    3    3    4 0
IRF     2562      1.69E+13    3    3    4 0
IRF     2642      1.69E+13    3    3    4 0
IRF     2722      1.70E+13    3    3    4 0
IRF     2802      1.70E+13    3    3    4 0
IRF     2882      1.70E+13    3    3    4 0
IRF     2962      1.70E+13    3    3    4 0
IRF     3042      1.71E+13    3    4    4 0
```

```
    IRF      3122     1.71E+13     4    4     4 0
    IRF      3202     1.71E+13     4    4     4 0
    IRF      3282     1.71E+13     4    4     4 0
    IRF      3362     1.71E+13     4    4     4 0
    IRF      3442     1.72E+13     4    4     4 0
    IRF      3522     1.72E+13     4    4     4 0
    IRF      3602     1.72E+13     4    4     4 0
    IRF      3682     1.73E+13     4    4     4 0
    IRF      3762     1.73E+13     4    5     4 0
    IRF      3842     1.73E+13     5    5     4 0
    IRF      3922     1.73E+13     5    5     4 0
    IRF      4002     1.74E+13     5    5     4 0
    IRF      4082     1.74E+13     5    5     4 0
    IRF      4162     1.74E+13     5    5     4 0
    IRF      4242     1.74E+13     5    5     4 0
    IRF      4322     1.75E+13     5    5     4 0
    IRF      4402     1.75E+13     5    5     4 0
    BUP
    RDA
    RDA      ***** OUTPUT MODULE *****
    TIT       CALDER HALL FUEL - STANDARD BURNUP
    BAS       1 MTIHM NAT U
    OPTL      8 8 8 8 5 19*8
    OPTA      8 8 8 8 5 19*8
    OPTF      8 8 8 8 5 19*8
    OUT       5    1      -1     0
    END
1   30060    2.339E-05     30070  3.408E-04     40090   1.000E-15     50100
2.948E-06
1   50110    1.305E-05     60120  9.874E+02     60130   1.201E+01    130270
6.998E-03
1  140280    7.348E-02    140290  3.853E-03    140300   2.646E-03    160320
2.367E-06
1  160330    1.927E-08    160340  1.114E-07    160360   5.605E-10    170350
1.4938E-09
1  170370    5.050E-10    200400  6.182E-06    200420   4.332E-08    200430
9.255E-09
1  200440    1.463E-07    200460  2.933E-10    200480   1.431E-08    220460
5.066E-09
1  220470    4.668E-09    220480  4.723E-08    220490   3.538E-09    220500
3.457E-09
1  230500    3.919E-09    230510  1.595E-06    240500   1.043E-03    240520
2.092E-02
1  240530    2.418E-03    240540  6.132E-04    260540   1.411E-03    260560
2.297E-02
1  260570    5.399E-04    260580  7.311E-05    280580   4.031E-03    280600
1.606E-03
1  280610    7.099E-05    280620  2.300E-04    280640   6.050E-05    300640
7.595E-11
1  300660    4.496E-11    300670  6.708E-12    300680   3.122E-11    300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03        0
0.0
 0
```

One-Fourth Initial Titanium Concentration

```
 -1
 -1
 -1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                         VAR. XSECT
LIB      0    1 2 3     922 923 924    9   50   0  1   0
INP      1  1  -1  -1   1  1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF         1     1.67E+13     1    2    4 2
IRF         2     1.68E+13     2    2    4 0
IRF        82     1.68E+13     2    2    4 0
IRF       162     1.68E+13     2    2    4 0
IRF       242     1.67E+13     2    2    4 0
IRF       322     1.67E+13     2    2    4 0
IRF       402     1.67E+13     2    2    4 0
IRF       482     1.66E+13     2    2    4 0
IRF       562     1.66E+13     2    2    4 0
IRF       642     1.66E+13     2    2    4 0
IRF       722     1.66E+13     2    2    4 0
IRF       802     1.66E+13     2    2    4 0
IRF       882     1.66E+13     2    2    4 0
IRF       962     1.65E+13     2    2    4 0
IRF      1042     1.66E+13     2    2    4 0
IRF      1122     1.65E+13     2    2    4 0
IRF      1202     1.66E+13     2    2    4 0
IRF      1282     1.66E+13     2    2    4 0
IRF      1362     1.66E+13     2    2    4 0
IRF      1442     1.66E+13     2    2    4 0
IRF      1522     1.66E+13     2    2    4 0
IRF      1602     1.66E+13     2    2    4 0
IRF      1682     1.66E+13     2    2    4 0
IRF      1762     1.66E+13     2    2    4 0
IRF      1842     1.67E+13     2    2    4 0
IRF      1922     1.67E+13     2    2    4 0
IRF      2002     1.67E+13     2    2    4 0
IRF      2082     1.67E+13     2    2    4 0
IRF      2162     1.68E+13     2    2    4 0
IRF      2242     1.68E+13     2    2    4 0
IRF      2322     1.68E+13     2    3    4 0
IRF      2402     1.69E+13     3    3    4 0
IRF      2482     1.69E+13     3    3    4 0
IRF      2562     1.69E+13     3    3    4 0
IRF      2642     1.69E+13     3    3    4 0
IRF      2722     1.70E+13     3    3    4 0
IRF      2802     1.70E+13     3    3    4 0
IRF      2882     1.70E+13     3    3    4 0
IRF      2962     1.70E+13     3    3    4 0
IRF      3042     1.71E+13     3    4    4 0
```

```
IRF      3122      1.71E+13      4    4    4 0
IRF      3202      1.71E+13      4    4    4 0
IRF      3282      1.71E+13      4    4    4 0
IRF      3362      1.71E+13      4    4    4 0
IRF      3442      1.72E+13      4    4    4 0
IRF      3522      1.72E+13      4    4    4 0
IRF      3602      1.72E+13      4    4    4 0
IRF      3682      1.73E+13      4    4    4 0
IRF      3762      1.73E+13      4    5    4 0
IRF      3842      1.73E+13      5    5    4 0
IRF      3922      1.73E+13      5    5    4 0
IRF      4002      1.74E+13      5    5    4 0
IRF      4082      1.74E+13      5    5    4 0
IRF      4162      1.74E+13      5    5    4 0
IRF      4242      1.74E+13      5    5    4 0
IRF      4322      1.75E+13      5    5    4 0
IRF      4402      1.75E+13      5    5    4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT       CALDER HALL FUEL - STANDARD BURNUP
BAS       1 MTIHM NAT U
OPTL       8 8 8 8 5 19*8
OPTA       8 8 8 8 5 19*8
OPTF       8 8 8 8 5 19*8
OUT       5     1     -1     0
END
1  30060    2.339E-05    30070   3.408E-04    40090   1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120   9.874E+02    60130   1.201E+01    130270
6.998E-03
1  140280   7.348E-02    140290  3.853E-03    140300  2.646E-03    160320
2.367E-06
1  160330   1.927E-08    160340  1.114E-07    160360  5.605E-10    170350
7.469E-10
1  170370   2.525E-10    200400  6.182E-06    200420  4.332E-08    200430
9.255E-09
1  200440   1.463E-07    200460  2.933E-10    200480  1.431E-08    220460
1.2665E-09
1  220470   1.167E-09    220480  1.1808E-08   220490  8.845E-10    220500
8.6425E-10
1  230500   3.919E-09    230510  1.595E-06    240500  1.043E-03    240520
2.092E-02
1  240530   2.418E-03    240540  6.132E-04    260540  1.411E-03    260560
2.297E-02
1  260570   5.399E-04    260580  7.311E-05    280580  4.031E-03    280600
1.606E-03
1  280610   7.099E-05    280620  2.300E-04    280640  6.050E-05    300640
7.595E-11
1  300660   4.496E-11    300670  6.708E-12    300680  3.122E-11    300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03        0
0.0
 0
```

One-Half Initial Titanium Concentration

```
 -1
 -1
 -1
RDA  *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **   CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                              VAR. XSECT
LIB     0    1 2 3      922 923 924     9    50    0  1    0
INP     1  1  -1  -1   1   1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF         1      1.67E+13     1    2    4 2
IRF         2      1.68E+13     2    2    4 0
IRF        82      1.68E+13     2    2    4 0
IRF       162      1.68E+13     2    2    4 0
IRF       242      1.67E+13     2    2    4 0
IRF       322      1.67E+13     2    2    4 0
IRF       402      1.67E+13     2    2    4 0
IRF       482      1.66E+13     2    2    4 0
IRF       562      1.66E+13     2    2    4 0
IRF       642      1.66E+13     2    2    4 0
IRF       722      1.66E+13     2    2    4 0
IRF       802      1.66E+13     2    2    4 0
IRF       882      1.66E+13     2    2    4 0
IRF       962      1.65E+13     2    2    4 0
IRF      1042      1.66E+13     2    2    4 0
IRF      1122      1.65E+13     2    2    4 0
IRF      1202      1.66E+13     2    2    4 0
IRF      1282      1.66E+13     2    2    4 0
IRF      1362      1.66E+13     2    2    4 0
IRF      1442      1.66E+13     2    2    4 0
IRF      1522      1.66E+13     2    2    4 0
IRF      1602      1.66E+13     2    2    4 0
IRF      1682      1.66E+13     2    2    4 0
IRF      1762      1.66E+13     2    2    4 0
IRF      1842      1.67E+13     2    2    4 0
IRF      1922      1.67E+13     2    2    4 0
IRF      2002      1.67E+13     2    2    4 0
IRF      2082      1.67E+13     2    2    4 0
IRF      2162      1.68E+13     2    2    4 0
IRF      2242      1.68E+13     2    2    4 0
IRF      2322      1.68E+13     2    3    4 0
IRF      2402      1.69E+13     3    3    4 0
IRF      2482      1.69E+13     3    3    4 0
IRF      2562      1.69E+13     3    3    4 0
IRF      2642      1.69E+13     3    3    4 0
IRF      2722      1.70E+13     3    3    4 0
IRF      2802      1.70E+13     3    3    4 0
IRF      2882      1.70E+13     3    3    4 0
IRF      2962      1.70E+13     3    3    4 0
IRF      3042      1.71E+13     3    4    4 0
```

```
IRF      3122      1.71E+13     4    4     4 0
IRF      3202      1.71E+13     4    4     4 0
IRF      3282      1.71E+13     4    4     4 0
IRF      3362      1.71E+13     4    4     4 0
IRF      3442      1.72E+13     4    4     4 0
IRF      3522      1.72E+13     4    4     4 0
IRF      3602      1.72E+13     4    4     4 0
IRF      3682      1.73E+13     4    4     4 0
IRF      3762      1.73E+13     4    5     4 0
IRF      3842      1.73E+13     5    5     4 0
IRF      3922      1.73E+13     5    5     4 0
IRF      4002      1.74E+13     5    5     4 0
IRF      4082      1.74E+13     5    5     4 0
IRF      4162      1.74E+13     5    5     4 0
IRF      4242      1.74E+13     5    5     4 0
IRF      4322      1.75E+13     5    5     4 0
IRF      4402      1.75E+13     5    5     4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT       CALDER HALL FUEL - STANDARD BURNUP
BAS       1 MTIHM NAT U
OPTL       8 8 8 8 5 19*8
OPTA       8 8 8 8 5 19*8
OPTF       8 8 8 8 5 19*8
OUT        5    1     -1    0
END
1   30060    2.339E-05     30070   3.408E-04      40090  1.000E-15     50100
2.948E-06
1   50110    1.305E-05     60120   9.874E+02      60130  1.201E+01    130270
6.998E-03
1  140280    7.348E-02    140290   3.853E-03     140300  2.646E-03    160320
2.367E-06
1  160330    1.927E-08    160340   1.114E-07     160360  5.605E-10    170350
7.469E-10
1  170370    2.525E-10    200400   6.182E-06     200420  4.332E-08    200430
9.255E-09
1  200440    1.463E-07    200460   2.933E-10     200480  1.431E-08    220460
2.533E-09
1  220470    2.334E-09    220480  2.3615E-08     220490  1.769E-09    220500
1.7285E-09
1  230500    3.919E-09    230510   1.595E-06     240500  1.043E-03    240520
2.092E-02
1  240530    2.418E-03    240540   6.132E-04     260540  1.411E-03    260560
2.297E-02
1  260570    5.399E-04    260580   7.311E-05     280580  4.031E-03    280600
1.606E-03
1  280610    7.099E-05    280620   2.300E-04     280640  6.050E-05    300640
7.595E-11
1  300660    4.496E-11    300670   6.708E-12     300680  3.122E-11    300700
1.026E-12
```

```
1  561300   1.002E-05    561320  9.698E-06    561340  2.356E-04    561350
6.474E-04
1  561360   7.770E-04    561370  1.119E-03    561380  7.198E-03         0
0.0
 0
```

Double Initial Titanium Concentration

```
 -1
 -1
 -1
RDA  *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA  **   CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                          VAR. XSECT
LIB      0    1 2 3      922 923 924     9    50    0   1    0
INP      1   1  -1  -1   1   1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1     1.67E+13    1    2    4 2
IRF        2     1.68E+13    2    2    4 0
IRF       82     1.68E+13    2    2    4 0
IRF      162     1.68E+13    2    2    4 0
IRF      242     1.67E+13    2    2    4 0
IRF      322     1.67E+13    2    2    4 0
IRF      402     1.67E+13    2    2    4 0
IRF      482     1.66E+13    2    2    4 0
IRF      562     1.66E+13    2    2    4 0
IRF      642     1.66E+13    2    2    4 0
IRF      722     1.66E+13    2    2    4 0
IRF      802     1.66E+13    2    2    4 0
IRF      882     1.66E+13    2    2    4 0
IRF      962     1.65E+13    2    2    4 0
IRF     1042     1.66E+13    2    2    4 0
IRF     1122     1.65E+13    2    2    4 0
IRF     1202     1.66E+13    2    2    4 0
IRF     1282     1.66E+13    2    2    4 0
IRF     1362     1.66E+13    2    2    4 0
IRF     1442     1.66E+13    2    2    4 0
IRF     1522     1.66E+13    2    2    4 0
IRF     1602     1.66E+13    2    2    4 0
IRF     1682     1.66E+13    2    2    4 0
IRF     1762     1.66E+13    2    2    4 0
IRF     1842     1.67E+13    2    2    4 0
IRF     1922     1.67E+13    2    2    4 0
IRF     2002     1.67E+13    2    2    4 0
IRF     2082     1.67E+13    2    2    4 0
IRF     2162     1.68E+13    2    2    4 0
IRF     2242     1.68E+13    2    2    4 0
IRF     2322     1.68E+13    2    3    4 0
IRF     2402     1.69E+13    3    3    4 0
IRF     2482     1.69E+13    3    3    4 0
IRF     2562     1.69E+13    3    3    4 0
IRF     2642     1.69E+13    3    3    4 0
IRF     2722     1.70E+13    3    3    4 0
IRF     2802     1.70E+13    3    3    4 0
IRF     2882     1.70E+13    3    3    4 0
IRF     2962     1.70E+13    3    3    4 0
IRF     3042     1.71E+13    3    4    4 0
```

```
      IRF      3122     1.71E+13    4    4    4 0
      IRF      3202     1.71E+13    4    4    4 0
      IRF      3282     1.71E+13    4    4    4 0
      IRF      3362     1.71E+13    4    4    4 0
      IRF      3442     1.72E+13    4    4    4 0
      IRF      3522     1.72E+13    4    4    4 0
      IRF      3602     1.72E+13    4    4    4 0
      IRF      3682     1.73E+13    4    4    4 0
      IRF      3762     1.73E+13    4    5    4 0
      IRF      3842     1.73E+13    5    5    4 0
      IRF      3922     1.73E+13    5    5    4 0
      IRF      4002     1.74E+13    5    5    4 0
      IRF      4082     1.74E+13    5    5    4 0
      IRF      4162     1.74E+13    5    5    4 0
      IRF      4242     1.74E+13    5    5    4 0
      IRF      4322     1.75E+13    5    5    4 0
      IRF      4402     1.75E+13    5    5    4 0
      BUP
      RDA
      RDA      ***** OUTPUT MODULE *****
      TIT      CALDER HALL FUEL - STANDARD BURNUP
      BAS      1 MTIHM NAT U
      OPTL      8 8 8 8 5 19*8
      OPTA      8 8 8 8 5 19*8
      OPTF      8 8 8 8 5 19*8
      OUT      5    1    -1    0
      END
1  30060    2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120  9.874E+02    60130  1.201E+01   130270
6.998E-03
1  140280   7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
2.367E-06
1  160330   1.927E-08   160340  1.114E-07   160360  5.605E-10   170350
7.469E-10
1  170370   2.525E-10   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1  200440   1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
1.0132E-08
1  220470   9.336E-09   220480  9.446E-08   220490  7.076E-09   220500
6.914E-09
1  230500   3.919E-09   230510  1.595E-06   240500  1.043E-03   240520
2.092E-02
1  240530   2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1  260570   5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1  280610   7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1  300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1   561300   1.002E-05   561320   9.698E-06   561340   2.356E-04   561350
6.474E-04
1   561360   7.770E-04   561370   1.119E-03   561380   7.198E-03          0
0.0
 0
```

One-Fourth Initial Sulfur Concentration

```
 -1
 -1
 -1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA       DECAY LIB    XSECT LIB                          VAR. XSECT
LIB     0    1 2 3     922 923 924     9   50    0  1    0
INP     1  1  -1  -1   1  1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1    1.67E+13    1    2    4 2
IRF        2    1.68E+13    2    2    4 0
IRF       82    1.68E+13    2    2    4 0
IRF      162    1.68E+13    2    2    4 0
IRF      242    1.67E+13    2    2    4 0
IRF      322    1.67E+13    2    2    4 0
IRF      402    1.67E+13    2    2    4 0
IRF      482    1.66E+13    2    2    4 0
IRF      562    1.66E+13    2    2    4 0
IRF      642    1.66E+13    2    2    4 0
IRF      722    1.66E+13    2    2    4 0
IRF      802    1.66E+13    2    2    4 0
IRF      882    1.66E+13    2    2    4 0
IRF      962    1.65E+13    2    2    4 0
IRF     1042    1.66E+13    2    2    4 0
IRF     1122    1.65E+13    2    2    4 0
IRF     1202    1.66E+13    2    2    4 0
IRF     1282    1.66E+13    2    2    4 0
IRF     1362    1.66E+13    2    2    4 0
IRF     1442    1.66E+13    2    2    4 0
IRF     1522    1.66E+13    2    2    4 0
IRF     1602    1.66E+13    2    2    4 0
IRF     1682    1.66E+13    2    2    4 0
IRF     1762    1.66E+13    2    2    4 0
IRF     1842    1.67E+13    2    2    4 0
IRF     1922    1.67E+13    2    2    4 0
IRF     2002    1.67E+13    2    2    4 0
IRF     2082    1.67E+13    2    2    4 0
IRF     2162    1.68E+13    2    2    4 0
IRF     2242    1.68E+13    2    2    4 0
IRF     2322    1.68E+13    2    3    4 0
IRF     2402    1.69E+13    3    3    4 0
IRF     2482    1.69E+13    3    3    4 0
IRF     2562    1.69E+13    3    3    4 0
IRF     2642    1.69E+13    3    3    4 0
IRF     2722    1.70E+13    3    3    4 0
IRF     2802    1.70E+13    3    3    4 0
IRF     2882    1.70E+13    3    3    4 0
IRF     2962    1.70E+13    3    3    4 0
IRF     3042    1.71E+13    3    4    4 0
```

```
IRF      3122      1.71E+13     4    4    4 0
IRF      3202      1.71E+13     4    4    4 0
IRF      3282      1.71E+13     4    4    4 0
IRF      3362      1.71E+13     4    4    4 0
IRF      3442      1.72E+13     4    4    4 0
IRF      3522      1.72E+13     4    4    4 0
IRF      3602      1.72E+13     4    4    4 0
IRF      3682      1.73E+13     4    4    4 0
IRF      3762      1.73E+13     4    5    4 0
IRF      3842      1.73E+13     5    5    4 0
IRF      3922      1.73E+13     5    5    4 0
IRF      4002      1.74E+13     5    5    4 0
IRF      4082      1.74E+13     5    5    4 0
IRF      4162      1.74E+13     5    5    4 0
IRF      4242      1.74E+13     5    5    4 0
IRF      4322      1.75E+13     5    5    4 0
IRF      4402      1.75E+13     5    5    4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT      CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT      5    1     -1     0
END
1  30060    2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120  9.874E+02    60130  1.201E+01   130270
6.998E-03
1  140280   7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
5.918E-07
1  160330   4.818E-09   160340  2.785E-08   160360  1.401E-10   170350
7.469E-10
1  170370   2.525E-10   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1  200440   1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
5.066E-09
1  220470   4.668E-09   220480  4.723E-08   220490  3.538E-09   220500
3.457E-09
1  230500   3.919E-09   230510  1.595E-06   240500  1.043E-03   240520
2.092E-02
1  240530   2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1  260570   5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1  280610   7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1  300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03         0
0.0
 0
```

One-Half Initial Sulfur Concentration

```
 -1
 -1
 -1
 RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
 RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
 RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
 LIB     0    1 2 3     922 923 924    9   50   0  1   0
 INP     1  1  -1  -1   1  1
 TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
 BUP
 IRF        1     1.67E+13    1    2    4 2
 IRF        2     1.68E+13    2    2    4 0
 IRF       82     1.68E+13    2    2    4 0
 IRF      162     1.68E+13    2    2    4 0
 IRF      242     1.67E+13    2    2    4 0
 IRF      322     1.67E+13    2    2    4 0
 IRF      402     1.67E+13    2    2    4 0
 IRF      482     1.66E+13    2    2    4 0
 IRF      562     1.66E+13    2    2    4 0
 IRF      642     1.66E+13    2    2    4 0
 IRF      722     1.66E+13    2    2    4 0
 IRF      802     1.66E+13    2    2    4 0
 IRF      882     1.66E+13    2    2    4 0
 IRF      962     1.65E+13    2    2    4 0
 IRF     1042     1.66E+13    2    2    4 0
 IRF     1122     1.65E+13    2    2    4 0
 IRF     1202     1.66E+13    2    2    4 0
 IRF     1282     1.66E+13    2    2    4 0
 IRF     1362     1.66E+13    2    2    4 0
 IRF     1442     1.66E+13    2    2    4 0
 IRF     1522     1.66E+13    2    2    4 0
 IRF     1602     1.66E+13    2    2    4 0
 IRF     1682     1.66E+13    2    2    4 0
 IRF     1762     1.66E+13    2    2    4 0
 IRF     1842     1.67E+13    2    2    4 0
 IRF     1922     1.67E+13    2    2    4 0
 IRF     2002     1.67E+13    2    2    4 0
 IRF     2082     1.67E+13    2    2    4 0
 IRF     2162     1.68E+13    2    2    4 0
 IRF     2242     1.68E+13    2    2    4 0
 IRF     2322     1.68E+13    2    3    4 0
 IRF     2402     1.69E+13    3    3    4 0
 IRF     2482     1.69E+13    3    3    4 0
 IRF     2562     1.69E+13    3    3    4 0
 IRF     2642     1.69E+13    3    3    4 0
 IRF     2722     1.70E+13    3    3    4 0
 IRF     2802     1.70E+13    3    3    4 0
 IRF     2882     1.70E+13    3    3    4 0
 IRF     2962     1.70E+13    3    3    4 0
 IRF     3042     1.71E+13    3    4    4 0
```

```
      IRF       3122      1.71E+13       4     4      4 0
      IRF       3202      1.71E+13       4     4      4 0
      IRF       3282      1.71E+13       4     4      4 0
      IRF       3362      1.71E+13       4     4      4 0
      IRF       3442      1.72E+13       4     4      4 0
      IRF       3522      1.72E+13       4     4      4 0
      IRF       3602      1.72E+13       4     4      4 0
      IRF       3682      1.73E+13       4     4      4 0
      IRF       3762      1.73E+13       4     5      4 0
      IRF       3842      1.73E+13       5     5      4 0
      IRF       3922      1.73E+13       5     5      4 0
      IRF       4002      1.74E+13       5     5      4 0
      IRF       4082      1.74E+13       5     5      4 0
      IRF       4162      1.74E+13       5     5      4 0
      IRF       4242      1.74E+13       5     5      4 0
      IRF       4322      1.75E+13       5     5      4 0
      IRF       4402      1.75E+13       5     5      4 0
      BUP
      RDA
      RDA     ***** OUTPUT MODULE *****
      TIT      CALDER HALL FUEL - STANDARD BURNUP
      BAS      1 MTIHM NAT U
      OPTL      8 8 8 8 5 19*8
      OPTA      8 8 8 8 5 19*8
      OPTF      8 8 8 8 5 19*8
      OUT       5     1     -1      0
      END
1   30060    2.339E-05     30070   3.408E-04      40090   1.000E-15     50100
2.948E-06
1   50110    1.305E-05     60120   9.874E+02      60130   1.201E+01    130270
6.998E-03
1  140280    7.348E-02    140290   3.853E-03     140300   2.646E-03    160320
1.184E-06
1  160330    9.635E-09    160340   5.570E-08     160360   2.803E-10    170350
7.469E-10
1  170370    2.525E-10    200400   6.182E-06     200420   4.332E-08    200430
9.255E-09
1  200440    1.463E-07    200460   2.933E-10     200480   1.431E-08    220460
5.066E-09
1  220470    4.668E-09    220480   4.723E-08     220490   3.538E-09    220500
3.457E-09
1  230500    3.919E-09    230510   1.595E-06     240500   1.043E-03    240520
2.092E-02
1  240530    2.418E-03    240540   6.132E-04     260540   1.411E-03    260560
2.297E-02
1  260570    5.399E-04    260580   7.311E-05     280580   4.031E-03    280600
1.606E-03
1  280610    7.099E-05    280620   2.300E-04     280640   6.050E-05    300640
7.595E-11
1  300660    4.496E-11    300670   6.708E-12     300680   3.122E-11    300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03        0
0.0
 0
```

Double Initial Sulfur Concentration

```
-1
-1
-1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                          VAR. XSECT
LIB     0    1 2 3     922 923 924    9   50    0  1   0
INP     1  1  -1  -1   1   1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1    1.67E+13    1    2    4 2
IRF        2    1.68E+13    2    2    4 0
IRF       82    1.68E+13    2    2    4 0
IRF      162    1.68E+13    2    2    4 0
IRF      242    1.67E+13    2    2    4 0
IRF      322    1.67E+13    2    2    4 0
IRF      402    1.67E+13    2    2    4 0
IRF      482    1.66E+13    2    2    4 0
IRF      562    1.66E+13    2    2    4 0
IRF      642    1.66E+13    2    2    4 0
IRF      722    1.66E+13    2    2    4 0
IRF      802    1.66E+13    2    2    4 0
IRF      882    1.66E+13    2    2    4 0
IRF      962    1.65E+13    2    2    4 0
IRF     1042    1.66E+13    2    2    4 0
IRF     1122    1.65E+13    2    2    4 0
IRF     1202    1.66E+13    2    2    4 0
IRF     1282    1.66E+13    2    2    4 0
IRF     1362    1.66E+13    2    2    4 0
IRF     1442    1.66E+13    2    2    4 0
IRF     1522    1.66E+13    2    2    4 0
IRF     1602    1.66E+13    2    2    4 0
IRF     1682    1.66E+13    2    2    4 0
IRF     1762    1.66E+13    2    2    4 0
IRF     1842    1.67E+13    2    2    4 0
IRF     1922    1.67E+13    2    2    4 0
IRF     2002    1.67E+13    2    2    4 0
IRF     2082    1.67E+13    2    2    4 0
IRF     2162    1.68E+13    2    2    4 0
IRF     2242    1.68E+13    2    2    4 0
IRF     2322    1.68E+13    2    3    4 0
IRF     2402    1.69E+13    3    3    4 0
IRF     2482    1.69E+13    3    3    4 0
IRF     2562    1.69E+13    3    3    4 0
IRF     2642    1.69E+13    3    3    4 0
IRF     2722    1.70E+13    3    3    4 0
IRF     2802    1.70E+13    3    3    4 0
IRF     2882    1.70E+13    3    3    4 0
IRF     2962    1.70E+13    3    3    4 0
IRF     3042    1.71E+13    3    4    4 0
```

```
IRF      3122      1.71E+13    4    4    4 0
IRF      3202      1.71E+13    4    4    4 0
IRF      3282      1.71E+13    4    4    4 0
IRF      3362      1.71E+13    4    4    4 0
IRF      3442      1.72E+13    4    4    4 0
IRF      3522      1.72E+13    4    4    4 0
IRF      3602      1.72E+13    4    4    4 0
IRF      3682      1.73E+13    4    4    4 0
IRF      3762      1.73E+13    4    5    4 0
IRF      3842      1.73E+13    5    5    4 0
IRF      3922      1.73E+13    5    5    4 0
IRF      4002      1.74E+13    5    5    4 0
IRF      4082      1.74E+13    5    5    4 0
IRF      4162      1.74E+13    5    5    4 0
IRF      4242      1.74E+13    5    5    4 0
IRF      4322      1.75E+13    5    5    4 0
IRF      4402      1.75E+13    5    5    4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT      CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT      5    1     -1     0
END
1  30060   2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110   1.305E-05    60120  9.874E+02    60130  1.201E+01    130270
6.998E-03
1  140280  7.348E-02    140290 3.853E-03    140300 2.646E-03    160320
4.734E-06
1  160330  3.854E-08    160340 2.228E-07    160360 1.121E-09    170350
7.469E-10
1  170370  2.525E-10    200400 6.182E-06    200420 4.332E-08    200430
9.255E-09
1  200440  1.463E-07    200460 2.933E-10    200480 1.431E-08    220460
5.066E-09
1  220470  4.668E-09    220480 4.723E-08    220490 3.538E-09    220500
3.457E-09
1  230500  3.919E-09    230510 1.595E-06    240500 1.043E-03    240520
2.092E-02
1  240530  2.418E-03    240540 6.132E-04    260540 1.411E-03    260560
2.297E-02
1  260570  5.399E-04    260580 7.311E-05    280580 4.031E-03    280600
1.606E-03
1  280610  7.099E-05    280620 2.300E-04    280640 6.050E-05    300640
7.595E-11
1  300660  4.496E-11    300670 6.708E-12    300680 3.122E-11    300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03         0
0.0
 0
```

One-Fourth Initial Lithium Concentration

```
 -1
 -1
 -1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA       DECAY LIB   XSECT LIB                        VAR. XSECT
LIB      0    1 2 3      922 923 924     9    50    0  1    0
INP      1  1  -1  -1   1  1
TIT      FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1     1.67E+13    1    2    4 2
IRF        2     1.68E+13    2    2    4 0
IRF       82     1.68E+13    2    2    4 0
IRF      162     1.68E+13    2    2    4 0
IRF      242     1.67E+13    2    2    4 0
IRF      322     1.67E+13    2    2    4 0
IRF      402     1.67E+13    2    2    4 0
IRF      482     1.66E+13    2    2    4 0
IRF      562     1.66E+13    2    2    4 0
IRF      642     1.66E+13    2    2    4 0
IRF      722     1.66E+13    2    2    4 0
IRF      802     1.66E+13    2    2    4 0
IRF      882     1.66E+13    2    2    4 0
IRF      962     1.65E+13    2    2    4 0
IRF     1042     1.66E+13    2    2    4 0
IRF     1122     1.65E+13    2    2    4 0
IRF     1202     1.66E+13    2    2    4 0
IRF     1282     1.66E+13    2    2    4 0
IRF     1362     1.66E+13    2    2    4 0
IRF     1442     1.66E+13    2    2    4 0
IRF     1522     1.66E+13    2    2    4 0
IRF     1602     1.66E+13    2    2    4 0
IRF     1682     1.66E+13    2    2    4 0
IRF     1762     1.66E+13    2    2    4 0
IRF     1842     1.67E+13    2    2    4 0
IRF     1922     1.67E+13    2    2    4 0
IRF     2002     1.67E+13    2    2    4 0
IRF     2082     1.67E+13    2    2    4 0
IRF     2162     1.68E+13    2    2    4 0
IRF     2242     1.68E+13    2    2    4 0
IRF     2322     1.68E+13    2    3    4 0
IRF     2402     1.69E+13    3    3    4 0
IRF     2482     1.69E+13    3    3    4 0
IRF     2562     1.69E+13    3    3    4 0
IRF     2642     1.69E+13    3    3    4 0
IRF     2722     1.70E+13    3    3    4 0
IRF     2802     1.70E+13    3    3    4 0
IRF     2882     1.70E+13    3    3    4 0
IRF     2962     1.70E+13    3    3    4 0
IRF     3042     1.71E+13    3    4    4 0
```

```
IRF      3122     1.71E+13     4    4     4 0
IRF      3202     1.71E+13     4    4     4 0
IRF      3282     1.71E+13     4    4     4 0
IRF      3362     1.71E+13     4    4     4 0
IRF      3442     1.72E+13     4    4     4 0
IRF      3522     1.72E+13     4    4     4 0
IRF      3602     1.72E+13     4    4     4 0
IRF      3682     1.73E+13     4    4     4 0
IRF      3762     1.73E+13     4    5     4 0
IRF      3842     1.73E+13     5    5     4 0
IRF      3922     1.73E+13     5    5     4 0
IRF      4002     1.74E+13     5    5     4 0
IRF      4082     1.74E+13     5    5     4 0
IRF      4162     1.74E+13     5    5     4 0
IRF      4242     1.74E+13     5    5     4 0
IRF      4322     1.75E+13     5    5     4 0
IRF      4402     1.75E+13     5    5     4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT      CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT      5    1     -1    0
END
1  30060   5.848E-06    30070  8.520E-05    40090  1.000E-15    50100
2.948E-06
1  50110   1.305E-05    60120  9.874E+02    60130  1.201E+01   130270
6.998E-03
1  140280  7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
2.367E-06
1  160330  1.927E-08   160340  1.114E-07   160360  5.605E-10   170350
7.469E-10
1  170370  2.525E-10   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1  200440  1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
5.066E-09
1  220470  4.668E-09   220480  4.723E-08   220490  3.538E-09   220500
3.457E-09
1  230500  3.919E-09   230510  1.595E-06   240500  1.043E-03   240520
2.092E-02
1  240530  2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1  260570  5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1  280610  7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1  300660  4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1   561300   1.002E-05    561320   9.698E-06    561340   2.356E-04    561350
6.474E-04
1   561360   7.770E-04    561370   1.119E-03    561380   7.198E-03         0
0.0
 0
```

One-Half Initial Lithium Concentration

```
 -1
 -1
 -1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA         DECAY LIB   XSECT LIB                        VAR. XSECT
LIB     0    1 2 3      922 923 924    9   50   0  1   0
INP     1  1  -1  -1  1  1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1     1.67E+13    1    2    4 2
IRF        2     1.68E+13    2    2    4 0
IRF       82     1.68E+13    2    2    4 0
IRF      162     1.68E+13    2    2    4 0
IRF      242     1.67E+13    2    2    4 0
IRF      322     1.67E+13    2    2    4 0
IRF      402     1.67E+13    2    2    4 0
IRF      482     1.66E+13    2    2    4 0
IRF      562     1.66E+13    2    2    4 0
IRF      642     1.66E+13    2    2    4 0
IRF      722     1.66E+13    2    2    4 0
IRF      802     1.66E+13    2    2    4 0
IRF      882     1.66E+13    2    2    4 0
IRF      962     1.65E+13    2    2    4 0
IRF     1042     1.66E+13    2    2    4 0
IRF     1122     1.65E+13    2    2    4 0
IRF     1202     1.66E+13    2    2    4 0
IRF     1282     1.66E+13    2    2    4 0
IRF     1362     1.66E+13    2    2    4 0
IRF     1442     1.66E+13    2    2    4 0
IRF     1522     1.66E+13    2    2    4 0
IRF     1602     1.66E+13    2    2    4 0
IRF     1682     1.66E+13    2    2    4 0
IRF     1762     1.66E+13    2    2    4 0
IRF     1842     1.67E+13    2    2    4 0
IRF     1922     1.67E+13    2    2    4 0
IRF     2002     1.67E+13    2    2    4 0
IRF     2082     1.67E+13    2    2    4 0
IRF     2162     1.68E+13    2    2    4 0
IRF     2242     1.68E+13    2    2    4 0
IRF     2322     1.68E+13    2    3    4 0
IRF     2402     1.69E+13    3    3    4 0
IRF     2482     1.69E+13    3    3    4 0
IRF     2562     1.69E+13    3    3    4 0
IRF     2642     1.69E+13    3    3    4 0
IRF     2722     1.70E+13    3    3    4 0
IRF     2802     1.70E+13    3    3    4 0
IRF     2882     1.70E+13    3    3    4 0
IRF     2962     1.70E+13    3    3    4 0
IRF     3042     1.71E+13    3    4    4 0
```

```
IRF     3122     1.71E+13    4    4    4 0
IRF     3202     1.71E+13    4    4    4 0
IRF     3282     1.71E+13    4    4    4 0
IRF     3362     1.71E+13    4    4    4 0
IRF     3442     1.72E+13    4    4    4 0
IRF     3522     1.72E+13    4    4    4 0
IRF     3602     1.72E+13    4    4    4 0
IRF     3682     1.73E+13    4    4    4 0
IRF     3762     1.73E+13    4    5    4 0
IRF     3842     1.73E+13    5    5    4 0
IRF     3922     1.73E+13    5    5    4 0
IRF     4002     1.74E+13    5    5    4 0
IRF     4082     1.74E+13    5    5    4 0
IRF     4162     1.74E+13    5    5    4 0
IRF     4242     1.74E+13    5    5    4 0
IRF     4322     1.75E+13    5    5    4 0
IRF     4402     1.75E+13    5    5    4 0
BUP
RDA
RDA     ***** OUTPUT MODULE *****
TIT     CALDER HALL FUEL - STANDARD BURNUP
BAS     1 MTIHM NAT U
OPTL     8 8 8 8 5 19*8
OPTA     8 8 8 8 5 19*8
OPTF     8 8 8 8 5 19*8
OUT     5     1     -1     0
END
1  30060    1.170E-05    30070   1.704E-04    40090   1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120   9.874E+02    60130   1.201E+01    130270
6.998E-03
1  140280   7.348E-02    140290  3.853E-03    140300  2.646E-03    160320
2.367E-06
1  160330   1.927E-08    160340  1.114E-07    160360  5.605E-10    170350
7.469E-10
1  170370   2.525E-10    200400  6.182E-06    200420  4.332E-08    200430
9.255E-09
1  200440   1.463E-07    200460  2.933E-10    200480  1.431E-08    220460
5.066E-09
1  220470   4.668E-09    220480  4.723E-08    220490  3.538E-09    220500
3.457E-09
1  230500   3.919E-09    230510  1.595E-06    240500  1.043E-03    240520
2.092E-02
1  240530   2.418E-03    240540  6.132E-04    260540  1.411E-03    260560
2.297E-02
1  260570   5.399E-04    260580  7.311E-05    280580  4.031E-03    280600
1.606E-03
1  280610   7.099E-05    280620  2.300E-04    280640  6.050E-05    300640
7.595E-11
1  300660   4.496E-11    300670  6.708E-12    300680  3.122E-11    300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03        0
0.0
 0
```

Double Initial Lithium Concentration

```
 -1
 -1
 -1
RDA *  BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB    0    1 2 3     922 923 924    9   50   0  1   0
INP    1  1  -1  -1   1   1
TIT    FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF       1    1.67E+13    1   2    4 2
IRF       2    1.68E+13    2   2    4 0
IRF      82    1.68E+13    2   2    4 0
IRF     162    1.68E+13    2   2    4 0
IRF     242    1.67E+13    2   2    4 0
IRF     322    1.67E+13    2   2    4 0
IRF     402    1.67E+13    2   2    4 0
IRF     482    1.66E+13    2   2    4 0
IRF     562    1.66E+13    2   2    4 0
IRF     642    1.66E+13    2   2    4 0
IRF     722    1.66E+13    2   2    4 0
IRF     802    1.66E+13    2   2    4 0
IRF     882    1.66E+13    2   2    4 0
IRF     962    1.65E+13    2   2    4 0
IRF    1042    1.66E+13    2   2    4 0
IRF    1122    1.65E+13    2   2    4 0
IRF    1202    1.66E+13    2   2    4 0
IRF    1282    1.66E+13    2   2    4 0
IRF    1362    1.66E+13    2   2    4 0
IRF    1442    1.66E+13    2   2    4 0
IRF    1522    1.66E+13    2   2    4 0
IRF    1602    1.66E+13    2   2    4 0
IRF    1682    1.66E+13    2   2    4 0
IRF    1762    1.66E+13    2   2    4 0
IRF    1842    1.67E+13    2   2    4 0
IRF    1922    1.67E+13    2   2    4 0
IRF    2002    1.67E+13    2   2    4 0
IRF    2082    1.67E+13    2   2    4 0
IRF    2162    1.68E+13    2   2    4 0
IRF    2242    1.68E+13    2   2    4 0
IRF    2322    1.68E+13    2   3    4 0
IRF    2402    1.69E+13    3   3    4 0
IRF    2482    1.69E+13    3   3    4 0
IRF    2562    1.69E+13    3   3    4 0
IRF    2642    1.69E+13    3   3    4 0
IRF    2722    1.70E+13    3   3    4 0
IRF    2802    1.70E+13    3   3    4 0
IRF    2882    1.70E+13    3   3    4 0
IRF    2962    1.70E+13    3   3    4 0
IRF    3042    1.71E+13    3   4    4 0
```

```
IRF      3122     1.71E+13     4    4     4 0
IRF      3202     1.71E+13     4    4     4 0
IRF      3282     1.71E+13     4    4     4 0
IRF      3362     1.71E+13     4    4     4 0
IRF      3442     1.72E+13     4    4     4 0
IRF      3522     1.72E+13     4    4     4 0
IRF      3602     1.72E+13     4    4     4 0
IRF      3682     1.73E+13     4    4     4 0
IRF      3762     1.73E+13     4    5     4 0
IRF      3842     1.73E+13     5    5     4 0
IRF      3922     1.73E+13     5    5     4 0
IRF      4002     1.74E+13     5    5     4 0
IRF      4082     1.74E+13     5    5     4 0
IRF      4162     1.74E+13     5    5     4 0
IRF      4242     1.74E+13     5    5     4 0
IRF      4322     1.75E+13     5    5     4 0
IRF      4402     1.75E+13     5    5     4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT      CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT      5    1     -1     0
END
1  30060    4.678E-05    30070   6.816E-04    40090   1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120   9.874E+02    60130   1.201E+01   130270
6.998E-03
1 140280    7.348E-02   140290   3.853E-03   140300   2.646E-03   160320
2.367E-06
1 160330    1.927E-08   160340   1.114E-07   160360   5.605E-10   170350
7.469E-10
1 170370    2.525E-10   200400   6.182E-06   200420   4.332E-08   200430
9.255E-09
1 200440    1.463E-07   200460   2.933E-10   200480   1.431E-08   220460
5.066E-09
1 220470    4.668E-09   220480   4.723E-08   220490   3.538E-09   220500
3.457E-09
1 230500    3.919E-09   230510   1.595E-06   240500   1.043E-03   240520
2.092E-02
1 240530    2.418E-03   240540   6.132E-04   260540   1.411E-03   260560
2.297E-02
1 260570    5.399E-04   260580   7.311E-05   280580   4.031E-03   280600
1.606E-03
1 280610    7.099E-05   280620   2.300E-04   280640   6.050E-05   300640
7.595E-11
1 300660    4.496E-11   300670   6.708E-12   300680   3.122E-11   300700
1.026E-12
```

```
1  561300   1.002E-05    561320  9.698E-06    561340  2.356E-04    561350
6.474E-04
1  561360   7.770E-04    561370  1.119E-03    561380  7.198E-03         0
0.0
 0
```

One-Fourth Initial Vanadium Concentration

```
 -1
 -1
 -1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                          VAR. XSECT
LIB      0    1 2 3     922 923 924    9   50   0  1   0
INP      1  1  -1  -1   1  1
TIT      FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF         1    1.67E+13    1    2    4 2
IRF         2    1.68E+13    2    2    4 0
IRF        82    1.68E+13    2    2    4 0
IRF       162    1.68E+13    2    2    4 0
IRF       242    1.67E+13    2    2    4 0
IRF       322    1.67E+13    2    2    4 0
IRF       402    1.67E+13    2    2    4 0
IRF       482    1.66E+13    2    2    4 0
IRF       562    1.66E+13    2    2    4 0
IRF       642    1.66E+13    2    2    4 0
IRF       722    1.66E+13    2    2    4 0
IRF       802    1.66E+13    2    2    4 0
IRF       882    1.66E+13    2    2    4 0
IRF       962    1.65E+13    2    2    4 0
IRF      1042    1.66E+13    2    2    4 0
IRF      1122    1.65E+13    2    2    4 0
IRF      1202    1.66E+13    2    2    4 0
IRF      1282    1.66E+13    2    2    4 0
IRF      1362    1.66E+13    2    2    4 0
IRF      1442    1.66E+13    2    2    4 0
IRF      1522    1.66E+13    2    2    4 0
IRF      1602    1.66E+13    2    2    4 0
IRF      1682    1.66E+13    2    2    4 0
IRF      1762    1.66E+13    2    2    4 0
IRF      1842    1.67E+13    2    2    4 0
IRF      1922    1.67E+13    2    2    4 0
IRF      2002    1.67E+13    2    2    4 0
IRF      2082    1.67E+13    2    2    4 0
IRF      2162    1.68E+13    2    2    4 0
IRF      2242    1.68E+13    2    2    4 0
IRF      2322    1.68E+13    2    3    4 0
IRF      2402    1.69E+13    3    3    4 0
IRF      2482    1.69E+13    3    3    4 0
IRF      2562    1.69E+13    3    3    4 0
IRF      2642    1.69E+13    3    3    4 0
IRF      2722    1.70E+13    3    3    4 0
IRF      2802    1.70E+13    3    3    4 0
IRF      2882    1.70E+13    3    3    4 0
IRF      2962    1.70E+13    3    3    4 0
IRF      3042    1.71E+13    3    4    4 0
```

```
IRF      3122     1.71E+13     4    4    4 0
IRF      3202     1.71E+13     4    4    4 0
IRF      3282     1.71E+13     4    4    4 0
IRF      3362     1.71E+13     4    4    4 0
IRF      3442     1.72E+13     4    4    4 0
IRF      3522     1.72E+13     4    4    4 0
IRF      3602     1.72E+13     4    4    4 0
IRF      3682     1.73E+13     4    4    4 0
IRF      3762     1.73E+13     4    5    4 0
IRF      3842     1.73E+13     5    5    4 0
IRF      3922     1.73E+13     5    5    4 0
IRF      4002     1.74E+13     5    5    4 0
IRF      4082     1.74E+13     5    5    4 0
IRF      4162     1.74E+13     5    5    4 0
IRF      4242     1.74E+13     5    5    4 0
IRF      4322     1.75E+13     5    5    4 0
IRF      4402     1.75E+13     5    5    4 0
BUP
RDA
RDA      ***** OUTPUT MODULE *****
TIT        CALDER HALL FUEL - STANDARD BURNUP
BAS      1 MTIHM NAT U
OPTL      8 8 8 8 5 19*8
OPTA      8 8 8 8 5 19*8
OPTF      8 8 8 8 5 19*8
OUT       5    1     -1     0
END
1  30060    2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110    1.305E-05    60120  9.874E+02    60130  1.201E+01   130270
6.998E-03
1  140280   7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
2.367E-06
1  160330   1.927E-08   160340  1.114E-07   160360  5.605E-10   170350
7.469E-10
1  170370   2.525E-10   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1  200440   1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
5.066E-09
1  220470   4.668E-09   220480  4.723E-08   220490  3.538E-09   220500
3.457E-09
1  230500   9.798E-10   230510  3.988E-07   240500  1.043E-03   240520
2.092E-02
1  240530   2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1  260570   5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1  280610   7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1  300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1  561300  1.002E-05  561320  9.698E-06  561340  2.356E-04  561350
6.474E-04
1  561360  7.770E-04  561370  1.119E-03  561380  7.198E-03        0
0.0
 0
```

One-Half Initial Vanadium Concentration

```
 -1
 -1
 -1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     922 923 924     9   50   0  1   0
INP     1  1  -1  -1   1  1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1     1.67E+13     1    2    4 2
IRF        2     1.68E+13     2    2    4 0
IRF       82     1.68E+13     2    2    4 0
IRF      162     1.68E+13     2    2    4 0
IRF      242     1.67E+13     2    2    4 0
IRF      322     1.67E+13     2    2    4 0
IRF      402     1.67E+13     2    2    4 0
IRF      482     1.66E+13     2    2    4 0
IRF      562     1.66E+13     2    2    4 0
IRF      642     1.66E+13     2    2    4 0
IRF      722     1.66E+13     2    2    4 0
IRF      802     1.66E+13     2    2    4 0
IRF      882     1.66E+13     2    2    4 0
IRF      962     1.65E+13     2    2    4 0
IRF     1042     1.66E+13     2    2    4 0
IRF     1122     1.65E+13     2    2    4 0
IRF     1202     1.66E+13     2    2    4 0
IRF     1282     1.66E+13     2    2    4 0
IRF     1362     1.66E+13     2    2    4 0
IRF     1442     1.66E+13     2    2    4 0
IRF     1522     1.66E+13     2    2    4 0
IRF     1602     1.66E+13     2    2    4 0
IRF     1682     1.66E+13     2    2    4 0
IRF     1762     1.66E+13     2    2    4 0
IRF     1842     1.67E+13     2    2    4 0
IRF     1922     1.67E+13     2    2    4 0
IRF     2002     1.67E+13     2    2    4 0
IRF     2082     1.67E+13     2    2    4 0
IRF     2162     1.68E+13     2    2    4 0
IRF     2242     1.68E+13     2    2    4 0
IRF     2322     1.68E+13     2    3    4 0
IRF     2402     1.69E+13     3    3    4 0
IRF     2482     1.69E+13     3    3    4 0
IRF     2562     1.69E+13     3    3    4 0
IRF     2642     1.69E+13     3    3    4 0
IRF     2722     1.70E+13     3    3    4 0
IRF     2802     1.70E+13     3    3    4 0
IRF     2882     1.70E+13     3    3    4 0
IRF     2962     1.70E+13     3    3    4 0
IRF     3042     1.71E+13     3    4    4 0
```

```
 IRF       3122     1.71E+13    4    4    4 0
 IRF       3202     1.71E+13    4    4    4 0
 IRF       3282     1.71E+13    4    4    4 0
 IRF       3362     1.71E+13    4    4    4 0
 IRF       3442     1.72E+13    4    4    4 0
 IRF       3522     1.72E+13    4    4    4 0
 IRF       3602     1.72E+13    4    4    4 0
 IRF       3682     1.73E+13    4    4    4 0
 IRF       3762     1.73E+13    4    5    4 0
 IRF       3842     1.73E+13    5    5    4 0
 IRF       3922     1.73E+13    5    5    4 0
 IRF       4002     1.74E+13    5    5    4 0
 IRF       4082     1.74E+13    5    5    4 0
 IRF       4162     1.74E+13    5    5    4 0
 IRF       4242     1.74E+13    5    5    4 0
 IRF       4322     1.75E+13    5    5    4 0
 IRF       4402     1.75E+13    5    5    4 0
 BUP
 RDA
 RDA      ***** OUTPUT MODULE *****
 TIT       CALDER HALL FUEL - STANDARD BURNUP
 BAS       1 MTIHM NAT U
 OPTL       8 8 8 8 5 19*8
 OPTA       8 8 8 8 5 19*8
 OPTF       8 8 8 8 5 19*8
 OUT        5    1     -1     0
 END
1  30060   2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110   1.305E-05    60120  9.874E+02    60130  1.201E+01   130270
6.998E-03
1 140280   7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
2.367E-06
1 160330   1.927E-08   160340  1.114E-07   160360  5.605E-10   170350
7.469E-10
1 170370   2.525E-10   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1 200440   1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
5.066E-09
1 220470   4.668E-09   220480  4.723E-08   220490  3.538E-09   220500
3.457E-09
1 230500   1.960E-09   230510  7.975E-07   240500  1.043E-03   240520
2.092E-02
1 240530   2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1 260570   5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1 280610   7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1 300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1  561300  1.002E-05    561320  9.698E-06    561340  2.356E-04    561350
6.474E-04
1  561360  7.770E-04    561370  1.119E-03    561380  7.198E-03         0
0.0
 0
```

Double Initial Vanadium Concentration

```
-1
-1
-1
RDA *   BURNUP OF 1 MTU OF NATURAL U IN CALDER HALL REACTOR
RDA **  CROSS SECTION LIBRARY = STEP 30 TAPE 9
RDA        DECAY LIB    XSECT LIB                          VAR. XSECT
LIB     0    1 2 3     922 923 924    9   50   0  1   0
INP     1  1  -1  -1   1  1
TIT     FLUX IRRADIATION OF 1 KG OF GRAPHITE MODERATOR
BUP
IRF        1    1.67E+13    1   2    4 2
IRF        2    1.68E+13    2   2    4 0
IRF       82    1.68E+13    2   2    4 0
IRF      162    1.68E+13    2   2    4 0
IRF      242    1.67E+13    2   2    4 0
IRF      322    1.67E+13    2   2    4 0
IRF      402    1.67E+13    2   2    4 0
IRF      482    1.66E+13    2   2    4 0
IRF      562    1.66E+13    2   2    4 0
IRF      642    1.66E+13    2   2    4 0
IRF      722    1.66E+13    2   2    4 0
IRF      802    1.66E+13    2   2    4 0
IRF      882    1.66E+13    2   2    4 0
IRF      962    1.65E+13    2   2    4 0
IRF     1042    1.66E+13    2   2    4 0
IRF     1122    1.65E+13    2   2    4 0
IRF     1202    1.66E+13    2   2    4 0
IRF     1282    1.66E+13    2   2    4 0
IRF     1362    1.66E+13    2   2    4 0
IRF     1442    1.66E+13    2   2    4 0
IRF     1522    1.66E+13    2   2    4 0
IRF     1602    1.66E+13    2   2    4 0
IRF     1682    1.66E+13    2   2    4 0
IRF     1762    1.66E+13    2   2    4 0
IRF     1842    1.67E+13    2   2    4 0
IRF     1922    1.67E+13    2   2    4 0
IRF     2002    1.67E+13    2   2    4 0
IRF     2082    1.67E+13    2   2    4 0
IRF     2162    1.68E+13    2   2    4 0
IRF     2242    1.68E+13    2   2    4 0
IRF     2322    1.68E+13    2   3    4 0
IRF     2402    1.69E+13    3   3    4 0
IRF     2482    1.69E+13    3   3    4 0
IRF     2562    1.69E+13    3   3    4 0
IRF     2642    1.69E+13    3   3    4 0
IRF     2722    1.70E+13    3   3    4 0
IRF     2802    1.70E+13    3   3    4 0
IRF     2882    1.70E+13    3   3    4 0
IRF     2962    1.70E+13    3   3    4 0
IRF     3042    1.71E+13    3   4    4 0
```

```
    IRF     3122     1.71E+13    4    4    4 0
    IRF     3202     1.71E+13    4    4    4 0
    IRF     3282     1.71E+13    4    4    4 0
    IRF     3362     1.71E+13    4    4    4 0
    IRF     3442     1.72E+13    4    4    4 0
    IRF     3522     1.72E+13    4    4    4 0
    IRF     3602     1.72E+13    4    4    4 0
    IRF     3682     1.73E+13    4    4    4 0
    IRF     3762     1.73E+13    4    5    4 0
    IRF     3842     1.73E+13    5    5    4 0
    IRF     3922     1.73E+13    5    5    4 0
    IRF     4002     1.74E+13    5    5    4 0
    IRF     4082     1.74E+13    5    5    4 0
    IRF     4162     1.74E+13    5    5    4 0
    IRF     4242     1.74E+13    5    5    4 0
    IRF     4322     1.75E+13    5    5    4 0
    IRF     4402     1.75E+13    5    5    4 0
    BUP
    RDA
    RDA     ***** OUTPUT MODULE *****
    TIT     CALDER HALL FUEL - STANDARD BURNUP
    BAS     1 MTIHM NAT U
    OPTL     8 8 8 8 5 19*8
    OPTA     8 8 8 8 5 19*8
    OPTF     8 8 8 8 5 19*8
    OUT     5    1    -1    0
    END
1  30060   2.339E-05    30070  3.408E-04    40090  1.000E-15    50100
2.948E-06
1  50110   1.305E-05    60120  9.874E+02    60130  1.201E+01   130270
6.998E-03
1 140280   7.348E-02   140290  3.853E-03   140300  2.646E-03   160320
2.367E-06
1 160330   1.927E-08   160340  1.114E-07   160360  5.605E-10   170350
7.469E-10
1 170370   2.525E-10   200400  6.182E-06   200420  4.332E-08   200430
9.255E-09
1 200440   1.463E-07   200460  2.933E-10   200480  1.431E-08   220460
5.066E-09
1 220470   4.668E-09   220480  4.723E-08   220490  3.538E-09   220500
3.457E-09
1 230500   7.838E-09   230510  3.190E-06   240500  1.043E-03   240520
2.092E-02
1 240530   2.418E-03   240540  6.132E-04   260540  1.411E-03   260560
2.297E-02
1 260570   5.399E-04   260580  7.311E-05   280580  4.031E-03   280600
1.606E-03
1 280610   7.099E-05   280620  2.300E-04   280640  6.050E-05   300640
7.595E-11
1 300660   4.496E-11   300670  6.708E-12   300680  3.122E-11   300700
1.026E-12
```

```
1  561300  1.002E-05   561320  9.698E-06   561340  2.356E-04   561350
6.474E-04
1  561360  7.770E-04   561370  1.119E-03   561380  7.198E-03         0
0.0
 0
```

VITA

Name: Kristin Elaine Chesson

Address: Department of Nuclear Engineering, 3133 TAMU, College Station, TX
                 77843-3133

Email Address: kchesson@tamu.edu

Education: B.S., Nuclear Engineering, Texas A&M University, 2004