Simulation of In-Core Dose Rates for an Offline CANDU Reactor

by

Jordan Gilbert

Bachelor of Nuclear Engineering (Hons), University of Ontario Institute of Technology, 2013

A THESIS SUBMITTED IN PARTIAL FULFILLMENT OF THE REQUIREMENTS FOR THE DEGREE OF

Master of Applied Science

in

The Faculty of Energy Systems and Nuclear Science

Nuclear Engineering

Supervisor(s): Ed Waller, University of Ontario Institute of Technology
Scott Nokleby, University of Ontario Institute of Technology
Examining Board: Glenn Harvel, University of Ontario Institute of Technology
External Examiner: Emily Corcoran, Royal Military College of Canada

THE UNIVERSITY OF ONTARIO INSTITUTE OF TECHNOLOGY

April 2016

©Jordan Gilbert 2016
Abstract

This thesis describes the development of a Monte Carlo simulation to predict the dose rates that will be encountered by a novel robotic inspection system for the pressure tubes of an offline CANDU reactor. Simulations were performed using the Monte Carlo N-Particle (MCNP) radiation transport code, version 6.1. The radiation fields within the reactor, even when shut down, are very high, and can cause significant damage to certain structural components and the electronics of the inspection system. Given that the robotic system will rely heavily on electronics, it is important to know the dose rates that will be encountered, in order to estimate the component lifetimes. The MCNP simulation was developed and benchmarked against information obtained from Ontario Power Generation and the Canadian Nuclear Laboratories. The benchmarking showed a good match between the simulated values and the expected values. This simulation, coupled with the accompanying user interface, represent a tool in dose field prediction that is currently unavailable. Predicted dose rates for a postulated inspection at 7 days after shutdown, with 2.5 cm of tungsten shielding around the key components, would survive for approximately 7 hours in core. This is anticipated to be enough time to perform an inspection and shows that the use of this tool can aid in designing the new inspection system.
Acknowledgements

I would first like to thank my supervisors, Dr. Ed Waller and Dr. Scott Nokleby, for their ongoing support and guidance in the completion of this thesis.

I would also like to thank my lab mates for their help in debugging code, solving problems that arose at the most inopportune times, and for helping me keep hold of at least some part of my sanity when all seemed lost.

Additionally, I need to give a special thanks to Duncan Barber, who helped by supplying the fuel source term for this work. Without you this work would have been nearly impossible.

Thanks to my family for helping me get through this and putting up with me even when I was stressed.

And last but not least, thanks to Laura, for all of your help and support and for understanding when I couldn’t always be around.
# Table of Contents

Abstract ii

Acknowledgements iii

Table of Contents vii

List of Tables viii

List of Figures x

Glossary xi

1 Introduction 1

1.1 Background ................................................. 2

1.1.1 CANDU Reactor ........................................ 2

1.1.1.1 Main Reactor Components .......................... 4

1.1.2 Flaw Types .............................................. 9

1.1.2.1 Fretting .............................................. 9

1.1.2.2 Delayed Hydride Cracking .......................... 11

1.1.2.3 Irradiation Enhanced Deformation .................. 13

1.1.2.4 Flaw Interactions .................................... 15

1.1.3 Inspection Systems ..................................... 16

1.1.4 CIGAR .................................................. 18
1.1.4.1 In-Core Inspection Head .................................. 19
1.1.4.2 Drive Mechanism ........................................ 22
1.1.5 Robotics in Radiation Environments ....................... 24
  1.1.5.1 Fukushima Daiichi Post-Accident Inspection ........... 24
  1.1.5.2 Three Mile Island Post Accident Inspection .......... 24
  1.1.5.3 Cleaning Steam Generators ............................ 25
  1.1.5.4 Gantry Crane for Handling Used Fuel Casks ........... 25
  1.1.5.5 Spent Fuel Storage Inspection ......................... 26
  1.1.5.6 Used Fuel Reprocessing ................................ 26
  1.1.5.7 Inspection of Pressurized Water Reactors (PWRs) .... 26
  1.1.5.8 Summary ................................................. 27

1.2 Problem Statement ............................................. 27
  1.2.1 Improved Inspection System .............................. 28
  1.2.2 Objectives ................................................ 29

1.3 Scope .................................................................... 30

2 Radiation Effects .................................................. 31
  2.1 Radiation Interaction With Materials ......................... 32
    2.1.1 Neutron Interactions ................................... 32
    2.1.2 Gamma Interactions ..................................... 32
      2.1.2.1 Photoelectric Effect ............................... 34
      2.1.2.2 Compton Scattering ............................... 34
      2.1.2.3 Pair Production .................................... 34
      2.1.2.4 Secondary Electrons ............................... 35
    2.1.3 Passive Components ..................................... 35
      2.2.1 Metals ................................................. 35
      2.2.2 Polymers .............................................. 36
      2.2.3 Rubbers ............................................... 36
<table>
<thead>
<tr>
<th>Section</th>
<th>Title</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.3</td>
<td>Effects on Electronics</td>
<td>38</td>
</tr>
<tr>
<td>2.3.1</td>
<td>Total Ionizing Dose</td>
<td>39</td>
</tr>
<tr>
<td>2.3.2</td>
<td>Other Damage Mechanics</td>
<td>40</td>
</tr>
<tr>
<td>2.3.3</td>
<td>Radiation Hardness</td>
<td>42</td>
</tr>
<tr>
<td>3</td>
<td>Monte Carlo N-Particle Simulation</td>
<td>44</td>
</tr>
<tr>
<td>3.1</td>
<td>Monte Carlo Method</td>
<td>45</td>
</tr>
<tr>
<td>3.1.1</td>
<td>Direct Inversion</td>
<td>47</td>
</tr>
<tr>
<td>3.1.2</td>
<td>Acceptance/Rejection</td>
<td>48</td>
</tr>
<tr>
<td>3.1.3</td>
<td>Discrete Data</td>
<td>49</td>
</tr>
<tr>
<td>3.2</td>
<td>MCNP Codes</td>
<td>51</td>
</tr>
<tr>
<td>3.2.1</td>
<td>MCNP Input</td>
<td>51</td>
</tr>
<tr>
<td>3.2.2</td>
<td>Verification Cases</td>
<td>53</td>
</tr>
<tr>
<td>3.3</td>
<td>Model</td>
<td>53</td>
</tr>
<tr>
<td>3.3.1</td>
<td>Fuel Bundle</td>
<td>55</td>
</tr>
<tr>
<td>3.3.2</td>
<td>Fuel Channel Assembly</td>
<td>55</td>
</tr>
<tr>
<td>3.3.3</td>
<td>Side Structural Components</td>
<td>57</td>
</tr>
<tr>
<td>3.3.3.1</td>
<td>End Fitting</td>
<td>57</td>
</tr>
<tr>
<td>3.3.3.2</td>
<td>Shield Plug</td>
<td>59</td>
</tr>
<tr>
<td>3.3.3.3</td>
<td>Shield Wall</td>
<td>61</td>
</tr>
<tr>
<td>3.3.4</td>
<td>Lattice</td>
<td>63</td>
</tr>
<tr>
<td>3.3.5</td>
<td>Outer Components</td>
<td>64</td>
</tr>
<tr>
<td>3.3.6</td>
<td>Source Term</td>
<td>66</td>
</tr>
<tr>
<td>3.3.6.1</td>
<td>Activation Term</td>
<td>70</td>
</tr>
<tr>
<td>3.3.7</td>
<td>Tallies</td>
<td>72</td>
</tr>
<tr>
<td>3.3.7.1</td>
<td>Shielding Cases</td>
<td>74</td>
</tr>
</tbody>
</table>
List of Tables

2.1 Radiation Hardness of Various Robotic Components . . . . . . . . 43

3.1 Gamma Energy Emission Rates at Several Key Decay Times [46] . 68

4.1 Total Fuel Activity for Different Fuel Terms . . . . . . . . . . . . 82

5.1 Simulated and Benchmark Dose Rates . . . . . . . . . . . . . . 91
5.2 Sensitivity of Dose to Bundle Power . . . . . . . . . . . . . . . . 92
5.3 Estimated Dose Rates at Different Days After Shutdown . . . . . 94
5.4 Radiation Hardness of Various Robotic Components . . . . . . . 98
5.5 Average Mass Energy Absorption Coefficient Calculations . . . . 100
5.6 Time to Failure for Unshielded Components in Channel M13 . . . 100
5.7 Estimated Failure Time for Shielded Components . . . . . . . . . 102

A.1 OPG Technical Drawings Used [41] . . . . . . . . . . . . . . . . . . 112
List of Figures

1.1 World Power Production (Modified from [1]) ......................... 1
1.2 CANDU System Overview [5] ........................................ 3
1.3 CANDU Fuel Bundle [7] ............................................. 4
1.4 CANDU Fuel Channel Assembly [8] ................................ 6
1.5 CANDU 6 Full Reactor Assembly [8] ............................... 8
1.6 Bundle 13 Fretting [10] .............................................. 10
1.7 Delayed Hydride Cracking Mechanism, Reproduced from [12] .... 11
1.9 CANDU Pressure Tube Failure Mechanisms, Adapted from [13] ... 16
1.10 CIGAR Inspection Head, Source: OPG ............................. 19
1.11 CIGAR Inspection Head with Labeled Components [6] ........... 20
1.12 CIGAR Ultrasonic Flaw Cluster [6] .................................. 21
1.13 Darlington Normal Closure Plug ..................................... 23
1.14 Darlington Modified Closure Plug .................................... 23
2.1 Photon Cross Section in Iron for Different Interactions [22] ....... 33
2.2 Effects of Radiation on Various Polymers (Modified from [25]) .... 37
3.1 Acceptance/Rejection Method ........................................ 49
3.2 (a) Probability Density Function and (b) Cumulative Probability Distribution for Discrete Data ......................... 50
3.3 MCNP6 Energy Ranges .............................................. 52
<table>
<thead>
<tr>
<th>Section</th>
<th>Description</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>3.4</td>
<td>Full Reactor Model</td>
<td>54</td>
</tr>
<tr>
<td>3.5</td>
<td>3D Rendering of MCNP Model of CANDU Fuel Bundle</td>
<td>56</td>
</tr>
<tr>
<td>3.6</td>
<td>Fuel Channel Assembly</td>
<td>57</td>
</tr>
<tr>
<td>3.7</td>
<td>Section View of the End Fitting</td>
<td>58</td>
</tr>
<tr>
<td>3.8</td>
<td>3D Rendering of the Shield Plug</td>
<td>60</td>
</tr>
<tr>
<td>3.9</td>
<td>Section View of the Shield Plug</td>
<td>60</td>
</tr>
<tr>
<td>3.10</td>
<td>Shield Wall Single Channel Close-Up</td>
<td>62</td>
</tr>
<tr>
<td>3.11</td>
<td>Shield Wall with Multiple Channels</td>
<td>62</td>
</tr>
<tr>
<td>3.12</td>
<td>3x3 Lattice</td>
<td>64</td>
</tr>
<tr>
<td>3.13</td>
<td>Full Reactor Core Section View</td>
<td>65</td>
</tr>
<tr>
<td>3.14</td>
<td>Full Reactor Side View</td>
<td>65</td>
</tr>
<tr>
<td>3.15</td>
<td>Non-Shielded Tally Arrangement</td>
<td>76</td>
</tr>
<tr>
<td>3.16</td>
<td>Shielded Tally Arrangement with 0 cm Shielding</td>
<td>76</td>
</tr>
<tr>
<td>3.17</td>
<td>Shielded Tally Arrangement with 2 cm Shielding</td>
<td>76</td>
</tr>
<tr>
<td>4.1</td>
<td>Graphical User Interface</td>
<td>78</td>
</tr>
<tr>
<td>4.2</td>
<td>Closeup of Channel Select Box</td>
<td>80</td>
</tr>
<tr>
<td>4.3</td>
<td>Closeup of Tally Selection Interface</td>
<td>81</td>
</tr>
<tr>
<td>4.4</td>
<td>Closeup of Shielding Case Selection Interface</td>
<td>83</td>
</tr>
<tr>
<td>4.5</td>
<td>File Architecture of Created Directories</td>
<td>88</td>
</tr>
<tr>
<td>5.1</td>
<td>Estimated Dose Rate and Total Fuel Activity vs. Days Since Shutdown</td>
<td>94</td>
</tr>
<tr>
<td>5.2</td>
<td>Axail Distribution of Dose Rate from Fuel Along Channel M13</td>
<td>95</td>
</tr>
<tr>
<td>5.3</td>
<td>Dose Rates for Various Shielding Materials</td>
<td>97</td>
</tr>
</tbody>
</table>
Glossary

AECL  Atomic Energy of Canada Limited.

AFS  Abnormal Fuel Support.

BBL  Break Before Leak.

BLIP  Blister and Spacer Location Inspection with PIPE.

BM  Burnish Mark.

BWR  Boiling Water Reactor.

CCTV  Closed Circuit TV.

CIGAR  Channel Inspection and Gauging Apparatus for Reactors.

CSTS  Calandria Side Tube Sheet.

CSV  Comma Separated Value.

DCF  Dose Conversion Factor.

DHC  Delayed Hydride Cracking.

FBBPF  Fuel Bundle Bearing Pad Frets.

FRILS  Fret Replica Inspection Laser Scanner.
**FSTS** Fueling machine Side Tube Sheet.

**GUI** Graphical User Interface.

**ICRP** International Commission on Radiological Protection.

**LANL** Los Alamos National Laboratory.

**LBB** Leak Before Break.

**LOCA** Loss Of Coolant Accident.

**LWR** Light Water Reactor.

**LZC** Liquid Zone Controllers.

**MARS** Mechatronic and Robotic Systems.

**MCNP** Monte Carlo N-Particle.

**MCNPX** Monte Carlo N-Particle eXtended.

**MOS** Metal-Oxide-Semiconductors.

**NSREC** Nuclear and Space Radiation Effects Conference.

**OPG** Ontario Power Generation.

**PDF** Probability Density Function.

**PHT** Primary Heat Transport.

**PIPE** Packaged Inspection ProbE.

**PWR** Pressurized Water Reactor.

**SDS** Shut Down System.
SEP  Single Event Phenomena.

SEU  Single Event Upset.

SLAR  Spacer Location And Re-positioning.

SSC  Side Structural Component.

TFC  Tally Fluctuation Chart.

TID  Total Ionizing Dose.

UDM  Universal Drive Machine.
Chapter 1

Introduction

Nuclear power is one of the cleanest sources of electricity worldwide, and with more than 435 commercial reactors operating in 31 countries it contributes approximately 11% of the world’s electricity [1]. Canada is a leader in the nuclear industry, and Ontario specifically uses nuclear power to supply almost 60% of its electricity needs [2]. Figure 1.1 shows the different types of power generation used and their proportions globally.

Figure 1.1: World Power Production (Modified from [1])
Nuclear energy production is a growing and ever changing field where newer and safer technology is always being developed and implemented. Canada is no exception, and has been developing nuclear technology since 1941, becoming the second country in the world to control a nuclear fission inside of a reactor (1945) [3]. Canada has also developed its own unique reactor design, the CANadian Deuterium Uranium (CANDU) reactor which is used in Canada, as well as being used in several countries world wide, including Argentina, China, Korea, and Romania [3].

This chapter will discuss some background on CANDU reactor design, different flaws that the CANDU pressure tubes are susceptible to, their importance, and some of the current inspection systems that are used to detect these flaws. Following this background section will be the problem statement, objectives and the scope of this work.

1.1 Background

1.1.1 CANDU Reactor

There are several key features that distinguish the CANDU reactor from other reactors. The first of these features is the use of heavy water as both moderator and coolant, as compared to PWRs and Boiling Water Reactors (BWRs), which are classified as Light Water Reactors (LWRs). Along with the use of heavy water, the CANDU reactor uses natural uranium fuel. This is an important distinction as it means that the uranium does not need to be enriched prior to being converted into UO$_2$ fuel. The third defining feature for the CANDU reactor is the use of multiple horizontal pressure tubes, as opposed to a single, large pressure vessel found in LWRs. The pressure tubes are where the fuel is loaded into and where the coolant passes through to remove the heat from the fuel. These horizontal pressure tubes also allow for the
final defining feature of the CANDU reactor, the ability to refuel online. PWRs, and BWRs both use a refuelling system known as batch refuelling, where the reactor is shutdown and all of the fuel is removed at once and replaced with fresh fuel. The downside of this refuelling method is the need to use burnable poisons to limit the initial reactivity of the fresh fuel. CANDUs get around this issue by refuelling small amounts, on a regular basis, while the reactor is still running. This means that the fresh fuel being inserted can be compensated for by placing it in strategic locations near older, burnt fuel, which serves to mitigate the local reactivity change. [4]

The main purpose of the CANDU reactor, and any commercial reactor, is to generate electricity. This is done by using the heat generated during the fission process to heat the coolant water. In the case of the CANDU reactor, with its heavy water coolant, this hot coolant then passes through a heat exchanger, also called a steam generator, which transfers the heat to a light water secondary side, as shown in Figure 1.2. This two side design has the benefit of reducing the likelihood of a radioactive...
release to the environment because the light water, which becomes steam, was never in the reactor and was not in contact with any radioactive material. The light water on the secondary side is converted into steam and is then passed through a set of turbines where it deposits most of its energy before going to a condenser and back to the steam generator. The energy that was deposited into the turbines causes them to spin, which is then used to turn a generator and create electricity. Figure 1.2 shows the full system from reactor to electricity production. [4]

1.1.1.1 Main Reactor Components

As mentioned in the previous section, the CANDU reactor is unique in its design and construction. The first difference is the fuel bundle. The CANDU fuel bundle is very small relative to other reactors, approximately 50 cm long, 10 cm in diameter, and 24 kg total mass [4]. The fuel bundle is composed of a number of fuel pencils (elements), an end cap on both ends to hold the elements together, and bearing pads, which are what contact the pressure tubes. Figure 1.3 shows a standard 37 element fuel bundle, like what is used in modern CANDU reactors. CANDU reactors use either a 12 or a 13 fuel bundle layout per channel, depending on the reactor [6].
The fuel channel is what holds the fuel, as well as providing the pressure boundary for the Primary Heat Transport (PHT) system. This pressure boundary is vital because the CANDU reactor, unlike BWRs, keeps the moderator separate from the coolant. In addition to this separation there is also a large difference in the pressure and temperature which each system is kept at. The moderator system is kept at low pressures and low temperatures. The coolant is kept at much higher pressure, approximately 11 MPa [4] in the pressure tube, and temperatures above 250°C [4].

The components that make up this pressure boundary are the pressure tube and the end fittings, and Figure 1.4 shows how these and other components are connected. The pressure tube is where the fuel actually sits and on either end is an end fitting, which houses the shield plug as well as the closure plug, which can be removed to allow for online refuelling. The pressure tube is connected to the end fitting using a rolled joint, where the pressure tube is forced out and into groves on the inside of the end fitting lip.

In addition to the pressure tube there is also the calandria tube. The calandria tube is located around the pressure tube, and is connected at either end to the shield wall with a rolled joint similar to that used for the pressure tube. Between the pressure tube and the calandria tube is a CO₂ gap that acts as thermal insulation to separate the high temperature pressure tube and coolant from the low temperature moderator. In order to ensure that this gap between the tubes stays constant there are four garter springs spread along the length of the pressure tube. It is very important that this gap is maintained because if the pressure tube is allowed to come into contact with the calandria tube the temperature gradient will cause reactions that can lead to Delayed Hydride Cracking (DHC). This phenomenon will be discussed in further detail in Section 1.1.2.2. [4,6]
Figure 1.4: CANDU Fuel Channel Assembly [8]
These fuel channels are laid out into a lattice, with each channel being 28.6 cm, centre to centre, from surrounding channels [4]. The number of channels that are laid out is dependant on the exact reactor, however, for this work a 480 channel CANDU was used. Specifically, the model has been based developed based on specifications for the Darlington 934 MW CANDU reactor [9]. The space between the channels is primarily taken up by moderator however there are also a number of control mechanisms threaded through the space. These control mechanisms include Liquid Zone Controllers (LZC), adjusters, absorbers, and shut down rods, as well as the pipes that distribute the liquid poison for Shut Down System (SDS)2. Around the entire lattice is the calandria tank, which serves to contain the moderator. Finally, on either end of the reactor are the shield walls. These shield walls consist of two large steel plates with a central cavity filled with light water and carbon steel balls. The walls have pass-throughs for the end fittings, and combined with the shield plugs serve to provide biological shielding, thereby reducing the amount of radiation that escapes the reactor. Figure 1.5 shows the full assembly of the CANDU 6 reactor, including a cutaway to show the internal structures discussed in this section. [4,6]

CANDU reactors rely on the pressure tubes to contain the fuel and to act as a pressure boundary for the coolant and if one of these pressure tubes were to rupture it could lead to a serious accident, known as a Loss Of Coolant Accident (LOCA). LOCAs are considered to be one of the most serious accident scenarios for CANDU reactors and as such, regular inspection is necessary to ensure that the pressure tubes are not at risk. The following section will explain some of the most common flaw types, their development mechanisms, and their threats, followed by a section discussing the current inspection systems used to monitor these flaws.
Figure 1.5: CANDU 6 Full Reactor Assembly [8]
1.1.2 Flaw Types

There are three main flaw types that can threaten the integrity of CANDU pressure tubes, potentially leading to a LOCA. These three types are fretting, DHC and irradiation enhanced deformation. Each of these flaw types has a different mechanism by which it operates and thereby each poses a different threat. The following sub-sections will give a more detailed explanation of the three types of flaws.

1.1.2.1 Fretting

One of the most common flaws that is encountered is known as a fret mark. There are two main kinds of fretting: Fuel Bundle Bearing Pad Frets (FBBPF), and debris frets. FBBPF are caused by fuel bundles moving or vibrating, thereby causing the bearing pads to rub against certain sections of the pressure tubes. FBBPF are generally rectangular in shape, relatively shallow, and not a significant threat to the pressure tube integrity. There are, however, some situations where this fretting can be accelerated, thereby leading to a deeper and more serious fret. In reactors that use a 13 fuel bundle layout, such as the Darlington reactor which this work focuses on, the #13 bundle, which sits on the inlet rolled joint, can be caused to vibrate more vigorously than other bundles due to the turbulence from the inlet flow. This increased vibration can lead to accelerated fretting on or near the rolled joint. Additionally, a phenomenon called Abnormal Fuel Support (AFS) can occur, where the fuel is not evenly supported, causing certain bearing pads to have uneven pressure with the pressure tube. This uneven pressure can lead to deep and extensive fretting in the Burnish Mark (BM) or mid plane areas, as shown in Figure 1.6. Either of these phenomenon can be a serious issue if not monitored and caught early. [6, 10]

The second type of fretting is known as debris fretting. This occurs when a piece of debris, generally foreign material, gets caught somewhere and cuts into the pressure
tube surface. The most common location for this is underneath a fuel bundle bearing pad. As the fuel vibrates in the flow, it presses the debris into the pressure tube. The debris can also cause damage during refueling movements, leading to long scratches. The size and shape of frets vary greatly depending on the piece of debris and where it gets caught. That being said, debris frets can be very deep and often have some sort of undercut due to the random motion of the debris in the flow. This can make the frets difficult to detect and potentially very dangerous, especially if there are sharp corners in the fret which could lead to higher than normal stress concentrations. [6,10]

In addition to FBBPF and debris fretting, there are several other mechanisms that create similar flaws. These include things like: refueling scratches, where a part of the fuel bundle scratches the surface of the pressure tube during refuelling; manufacturing flaws, which may be scratches and dents or may be other flaws; and crevice corrosion. Crevice corrosion is the most significant of these other flaws, and can be as significant as fretting. Crevice corrosion is caused when a localized boiling condition occurs where the bearing pads contact the pressure tube, leading to increased concentrations.
of LiOH. The number and severity of these crevices within the pressure tube are related to the temperature profile of the channel, and as such tend to cluster near the outlet, where the temperature is highest. These crevices are generally shallow and wide and seem to be self limiting in depth, and, as such, not a direct threat to pressure tube integrity. [6, 10, 11]

1.1.2.2 Delayed Hydride Cracking

DHC is a phenomenon that occurs in zirconium alloys such as those used in CANDU pressure tubes. DHC is a sub-critical growth mechanism, and involves the development of localized brittle hydride phases within the alloy structure. These localized areas of brittleness, combined with the stress on the pressure tube lead to the formation of a crack. Once the crack has begun, more hydride diffuses to the tip of the crack until it reaches a critical condition, which is dependant on the stress being applied. When this critical condition is reached, the stress causes the crack to grow, as shown in Figure 1.7. [12]

![Figure 1.7: Delayed Hydride Cracking Mechanism, Reproduced from [12]](image-url)
The growth rate of these cracks is highly dependant on the amount of hydrogen in solution in the zirconium alloy. There is always some amount of hydrogen in the zirconium alloy, however, the solubility limit is low. Hydrides can form using either hydrogen, from manufacturing, or deuterium, picked up during reactor operation. However, deuterium is only half as effective at forming hydrides as hydrogen is, and as such twice as much is needed [12]. The hydride concentration in the alloy is related to hydrogen/deuterium concentration, and to temperature. Hydrides are always present at room temperature, but do not form at high temperatures as easily. This mean that at reactor operating temperatures, higher hydrogen/deuterium concentrations are required for hydrides to form. As such, DHC can occur more easily when the reactor is cold, given that hydrides require less hydrogen/deuterium. Regardless of the temperature, sufficient localized hydride concentration and tensile stress is needed for DHC to occur. [11,12]

DHC is only a threat when there is hydrogen and deuterium present to form hydrides and, as such, there has been a great deal of work put into minimizing the amount of these elements in the pressure tubes. The amount of hydrogen present in the pressure tubes, which is introduced as a byproduct of manufacturing, has been reduced from original amounts of 5 to 25 ppm down to a maximum of 5 ppm. Deuterium ingress rates are generally around 1 ppm per year (hydrogen equivalent) and as such this decrease in the initial hydrogen increases the time needed to form hydrides by up to 20 years. This means that as long as conditions which cause localized buildups of hydrides are avoided, the pressure tubes should be safe throughout their 30 year lifetime. [11,12]

There are a couple different locations where localized concentrations of hydrides can become a problem. The first is when the pressure tube sags, due to either movement
of the garter springs or irradiation enhanced sag (as discussed in Section 1.1.2.3), and comes in contact with the cooler calandria tube. This contact, and the cold spot it makes on the pressure tube, can lead to the formation of a hydride blister. This is especially dangerous because, unlike most failures of the pressure tube which allow for Leak Before Break (LBB) detection methodology, hydride blisters can create a crack part of the way through the pressure tube meaning that LBB may not occur. This can happen when the garter springs move during construction of commissioning, and while this should not be a problem with the tight garter springs used in the newer reactors, inspections still attempt to localize the springs to be certain. The rolled joint area of the pressure tube is also at an increased risk of DHC because of the higher than normal deuterium ingress rates and higher tensile stresses. Finally, other flaws in the pressure tube, including frets, scratches, crevice corrosion locations, and any other surface flaw, can be locations where hydrides gather, leading to increased DHC chances. [6,11,12]

1.1.2.3 Irradiation Enhanced Deformation

Irradiation enhanced deformation is a phenomenon that affects the zirconium alloy that makes up CANDU pressure tubes. Pressure tubes are near-constantly exposed to high temperatures (∼300°C), high pressure (∼11 MPa) and high neutron fluxes (∼3.7 × 10^{13} n/cm^2/s) and fluences (∼3 × 10^{22} n/cm^2) [11], which lead to changes in the material. The primary deformations that occur are: sag, elongation, diametral expansion, and wall thinning. Figure 1.8 shows most of these deformations as compared to an original pressure tube. This section will discuss these different deformations, as well as their impact on the reactor, in more detail. [6,11]

Sag is the first form of irradiation enhanced deformation that will be discussed. Sag can happen in two different ways; the first being when the pressure tube sags between
the garter springs, and the second is when the entire length of the pressure tube sags. When the pressure tube sags between the garter springs it comes closer to the calandria tube, and, as discussed in Section 1.1.2.2, contact between these tubes is a very serious issue which can lead to the formation of a hydride blister, and may lead to a LOCA. This issue has been remedied in newer reactor designs by increasing the number of garter springs from the original two up to four, and by using tighter fitting springs that are much less likely to move. These improvements mean that there is little to no chance of a contact occurring, even well past the 30 year design lifetime of the pressure tubes. The second type of sag that occurs is over the full length of the pressure tube, and in this case the calandria tube will sag with it. For most channels this is not a concern however there are a number of pipes for SDS2 that run horizontally at 90° to the pressure tubes. Channels above these pipes could come into contact with the pipes, however this also should not occur in the design life. [6,11]

Over time, pressure tubes elongate as a result of the high temperature and fluxes that they are exposed to. This elongation is linear in nature, with peak growth rates of approximately 5 mm/year, and as such is easily predicted. This growth rate is however higher than the initial design expectations, and, as such, some early CANDUs,
while having some features to accommodate this elongation, do not have sufficient accommodations. All newer CANDUs includes several features which allow for this elongation to occur without causing issues over the entire design life. With these accommodations elongation does not present any serious concerns, however, inspections are still done to ensure that the growth rate is normal. [6, 11]

Diametral expansion is another type of irradiation enhanced deformation that affects CANDU pressure tubes. As the name implies, diametral expansion is the growth of the overall diameter of the pressure tube. This can become an issue due to the increased coolant flow that is allowed around the outside edge of the fuel bundles. When coolant flow is allowed to flow around the edge of the fuel it decreases the amount that flows through the bundle, thereby reducing the cooling that the centre fuel pencils receive. This is remedied by slightly increasing the flow rate as the reactor ages. The average growth rate for pressure tube diameter is around 0.1 mm/year, and is unlikely to ever reach the conservatively set limit of 5% increase in the initial diameter. [6, 11]

Diametral expansion can also lead to another side effect; wall thinning. As the diameter of the pressure tube expands, the walls can begin to thin. This wall thinning is well within the design expectations and as such this should not pose any significant threat to reactor operation or lifetime. Regular inspection monitors the wall thickness to ensure that there are no significant changes to the rate of thinning. [6, 11]

1.1.2.4 Flaw Interactions

As shown throughout this section, there are a number of different flaws that can occur within the CANDU pressure tubes. While some of these flaws can pose a threat to the pressure tube integrity, most have been designed for, and as such pose little to no threat. While one flaw alone may not pose a severe threat, the combination of several
Figure 1.9: CANDU Pressure Tube Failure Mechanisms, Adapted from [13]

key circumstances or multiple flaws can lead to a scenario where the pressure tube can be compromised. The CANDU has an additional safety feature in this case, the LBB system. This system detects any leak from pressure tube by constantly measuring the CO₂ annulus gas. This system means that if the pressure tube fails as it is meant to, it will leak and can be detected. This failure mode is not guaranteed however, and a Break Before Leak (BBL) failure can also occur. Figure 1.9 shows the failure mechanisms discussed throughout Section 1.1.2 and how they interact with each other to lead to a failure.

1.1.3 Inspection Systems

As discussed in Section 1.1.2, there are a number of different flaws that can occur in CANDU pressure tubes, however, most of them have been designed for and are not a threat to reactor safety as long as they are regularly inspected. These inspections are
carried out during regular outages, meaning that the reactor is in an offline condition. Being in an offline condition means that there will be a large amount of negative reactivity in the reactor core from all of the shutdown mechanisms. This means that any neutrons that are produced will have a very high chance of being absorbed, therefor limiting the number of fissions that can be caused. This means that delayed neutrons become the primary source of neutrons within the reactor, and after some time cooling, even these delayed neutrons will decrease. This means that the neutron flux is very low, low enough that it will be insignificant relative the amount of gamma radiation for the purpose of causing damage. The channel being inspected is defueled, and the shield plug is removed, leaving a the channel clear to be inspected by the inspection system which would be installed in the end fitting. There are several inspection systems that have been developed, as well as a number of custom built systems made for one specific inspection. Below is a list of some of the inspection systems that have been, or are currently used. [6,11]

**Dry Channel Gauging Equipment**

An early inspection system that measured sag, pressure tube diameter and inside tube profile.

**Rolled Joint Ultrasonic Inspection Equipment**

An early tool developed to inspect cracks in the rolled joint area.

**STEM Inspection Delivery Equipment**

An early eddy current and Closed Circuit TV (CCTV) inspection system.

**Packaged Inspection ProbE (PIPE)**

Designed to do fast ultrasonic inspections of rolled joint areas of pressure tubes, at rates of up to 10 channels (20 joints) per day.
Blister and Spacer Location Inspection with PIPE (BLIP)

Developed for the Pickering reactors in anticipation of the need to be able to locate spacers, and detect hydride blisters.

Scrape Tool

A tool used to remove small slivers of pressure tube material which could be tested to determine H/D ingress rates.

Spacer Location And Re-positioning (SLAR)

A tool designed to locate and, if need be, reposition garter springs for reactors fitted with the loose garter spring design.

Flaw Replicator

A tool that makes a silicon/rubber mold of flaws, thereby allowing for 3D laser scans to be made of the flaw using the Fret Replica Inspection Laser Scanner (FRILS) system. This is done on relatively large, potentially dangerous flaws found by other systems.

Channel Inspection and Gauging Apparatus for Reactors (CIGAR)

The CIGAR system is the currently used inspection system and is the main focus of this research. Given the key role it plays in this research, Section 1.1.4 will examine it in depth.

1.1.4 CIGAR

The CIGAR system is very different from most of the other inspection systems, in that it was designed to be a full channel inspection system. As opposed to tools made to only scan one part of the fuel channel, such as the rolled joint, the CIGAR does a volumetric inspection of the entire fuel channel. It does this with the use of a multitude of different ultrasonic, eddy current and other sensors. The inspection
head can be seen in Figure 1.10. In addition to being able to scan the entire volume of the pressure tube, the CIGAR system can also measure the pressure tube sag, gap between the pressure tube and calandria tube, and locate the garter springs. An added benefit of the CIGAR system over some earlier systems is that it requires minimal worker exposure due to the delivery system. It is this combination of diverse detection equipment, full volumetric scan capability, and low worker exposure which is why the CIGAR system is the primary inspection system used [11].

The CIGAR system is comprised of three main components: the inspection head, the drive mechanism and the computer controls. The computer control and data processing is done from a location outside of the containment building in order to reduce dose to workers. The cables are run through pass-throughs in the containment building, and into a local command node, before going to the drive mechanism and on to the inspection head. The inspection head will be described in Section 1.1.4.1, and the drive mechanism will be described in Section 1.1.4.2.

1.1.4.1 In-Core Inspection Head

The CIGAR inspection head is made up of a number of segments to allow for disassembly. There are five main segments that make up the inspection head, and these items can each be seen in Figure 1.11: the tailstock connector, labeled as HCR; the
universal joint, also called a constant velocity joint; a centering module; sag module; another centering module; and the ultrasonic module. The tailstock connector is what attaches the inspection head to the drive rod, which will be discussed in further detail in Section 1.1.4.2. The universal joint, often referred to simply as the U joint or CV joint, allows for the rotation from the drive rod to be passed smoothly to the inspection head regardless of tilt or centerline offset. The centering modules use spring loaded roller wheels to reduce the amount of friction on the pressure tube, thereby reducing the chance of damage occurring, by allowing the robot to rotate smoothly. Between the two centering modules is the sag module, where the sag sensors are stored. In addition to the sag sensors, the eddy current coil for garter spring detection is also mounted on this module, specifically around the outside. Finally is the sensor head, where a number of different ultrasonic sensors are held, in various arrangements. There are some newer models of the CIGAR inspection head which include an improved gap sensor between the rollers of the front centering module. [6,11]
The inspection head uses a number of different sensors to complete the full inspection, including: ultrasonic sensors, in various arrangements; eddy current coils; and a servo inclinometer, for sag measurement. The servo inclinometer takes sag profile measurements every 30 mm along the inspection which allows for an accurate representation of the sag within the pressure tube to be developed. There are two different eddy current coils that are used, the garter spring location coil, and the gap module, which is not shown. The garter spring module is primarily used on older reactors that are still using the loose spacers, however, attempts have been made to extend this use to tight spacers as well. The gap module, which is located between the rollers of the front centering module also uses eddy current to determine the space between the pressure tube and calandria tube. [6, 11, 14]

There are a number of different ultrasonic sensors, arranged in different orientations, all located within the ultrasonic module. The main set of ultrasonic sensors is referred to as the flaw cluster, and is shown in Figure 1.12 (the central set of five sensors)
along with some of the other ultrasonic sensors. This flaw cluster consists of four 10 MHz ultrasonic sensors, arranged in an angled configuration, along with a 20 MHz sensor, normal to the pressure tube surface. The four 10 MHz sensors work in pairs, one pair measuring circumferentially and one measuring axially, and, combined with the normal beam, are what perform the volumetric inspection, detecting any flaws above 0.15 mm deep. In addition to the flaw cluster, there is also a 10 MHz sensor mounted normal to the pressure tube, mounted 40 mm in front of the flaw cluster. This detector is what measures the diameter, as well as the wall thickness of the pressure tube. [6, 11, 14]

1.1.4.2 Drive Mechanism

The CIGAR head is driven through the pressure tube using a set of drive rods connected to an external drive unit. This method of driving means that there must be a way for the drive rods to pass through where the closure plug should be, and as such, requires the use of a modified closure plug. The drive system, or Universal Drive Machine (UDM), is either mounted to the fueling machine bridge or is delivered by a crane and can reach any given channel. The UDM is connected to the inspection head by the drive rods. These drive rods connect to the tailstock connector and serve to both drive the inspection head, both axially and rotationally, and to house the data transfer wires. There are two of these drive rods, a short rod and a long rod. These rods lock to the inspection head, and combined these allow for the entire pressure tube length to be inspected. As mentioned, these rods also carry the signals from the inspection head to the local console, where it is then sent to the command station. These signals are carried on a number of coaxial cables that are housed in the centre of the drive rods. These cables are connected to a slip ring at the UDM to ensure that they do not get tangled by the rotational drive. [6]
As noted earlier, the normal closure plug cannot be used due to the need for the drive rods to pass through, and as such a modified closure plug is used. These modified closure plugs vary depending on which reactor is being inspected, due to the different closure plug mechanisms used. Figure 1.13 and Figure 1.14 show the normal and modified closure plugs for the Darlington reactor, respectively. As mentioned in Section 1.1.3, the reactor is offline and cooled, however, there is still fuel in the reactor and as such coolant must be kept flowing. Reduced coolant temperatures and pressures are used during inspections, around 50°C and 262 kPa [15]. Due to this reduced pressure, the modified closure plug is sufficient to maintain the pressure boundary. This can become an issue though, if pressure needs to be increased for any reason, such as to maintain cooling. In these scenarios the drive mechanism is disconnected and the fueling machines are brought in to form the pressure boundary on the channel being inspected. [6]
1.1.5 Robotics in Radiation Environments

Robots are an essential component in many nuclear environments, allowing workers to do inspections or work in places where the radiation would normally prevent it. Beyond the inspection systems that have already been discussed there are a number of other applications for robotics in radiation environments. These applications range from robots to perform inspections after a nuclear accident, such as at the Fukushima Daiichi nuclear power plant, to gantry systems used in spent fuel storage facilities. This section will examine some of the other robots that are being used in radiation environments.

1.1.5.1 Fukushima Daiichi Post-Accident Inspection

In the wake of the events at the Fukushima Daiichi nuclear power plant in 2011 a number of radiation hardened robots have been sent into the destroyed reactor building in an attempt to locate the fuel rods, which have likely fallen to the bottom of the reactor vessel. On April 10, 2015, a robotic inspection system was inserted through a small pipe in order to look for access to the basement of the reactor building, such that an aquatic robot could then be sent to look for the fuel rods. The robot found the basement access was clear and continued its inspection before eventually getting stuck. During the inspection dose rates ranging from 7.4 Sv/hr to 9.7 Sv/hr were reported. While this dose rate is extremely dangerous to humans, with the $LD_{50}$ being approximately 4 Sv, these dose rates are reasonably low for inspection devices, provided they are designed for the environment and use hardened components. [16]

1.1.5.2 Three Mile Island Post Accident Inspection

In the same way that inspections were performed at Fukushima Daiichi after the accident, so to were inspections performed after the Three Mile Island accident in 1979. The difference between these inspections however is key. Where Fukushima
Daiichi had very limited access due to the structural damage of the building, Three Mile Island did not suffer the same damage, and as such, larger robots could be used. A number of robotic inspection systems were used to do various tasks, in various locations within the reactor building. The robots ranged from tracked inspection robots with cameras to large robots with arms capable of lifting 450 kg. One such inspection task was to monitor the demineralizer tank as the resins were flushed. The robot that performed this task was outfitted with radiation hardened electronics, including a live feed camera, and was receiving doses of 3,000 rad/hr (30 Gy/hr). [17]

1.1.5.3 Cleaning Steam Generators

Another application where robots have been used to limit worker exposure is for cleaning the steam generators at nuclear reactors. Cleaning and inspection of steam generators is done to ensure safety and to improve efficiency. Several robots have been designed for this application, and are outfitted with high pressure water jets to clean sludge which builds up. This sludge is radioactive and the steam generator has dose rates ranging from 50 mSv/hr to 250 mSv/hr. [18]

1.1.5.4 Gantry Crane for Handling Used Fuel Casks

Another common task for robots that involves radioactive environments is the moving of used fuel. This used fuel is generally stored in dry storage casks which help to limit the radiation fields around it. One such example is the work done by Sandia National Laboratories in which they analyzed the lifetime expectancy of such a gantry system. The proposed gantry system would be used in an interim storage facility that would house spent nuclear fuel while a disposal facility was prepared. Modeling was performed to estimate the dose to the robotic system and potential failures that could occur. The dose was found to be approximately 8 krad[Si] (80 Gy[Si]) per loading, resulting in a dose of 1.6 Mrad[Si] (16 kGy[Si]) per year assuming 200 loadings per
year. Another potential use of this same gantry system would be for decommissioning and decontamination of reactors. Simulations showed that the dose for this application would be approximately 35 rad[Si]/hr. Through these simulations, and a number of tests to determine the radiation hardness of limiting components, it was found that unless hardened components were used, the system would fail in less that one loading for the used fuel application. [19]

1.1.5.5 Spent Fuel Storage Inspection

Spent nuclear fuel is often stored in concrete casks until it can be permanently disposed of. These casks need to occasionally be inspected to ensure that they are still structurally sound and that no major cracking has occurred, especially near the end of their design life. These inspections are performed by a robotic inspection system equipped with a radiation hardened video camera. Dose rates within these casks vary, both based on the location within the cask and with time, but can exceed 2,000 R/hr (17.5 Gy/hr[Air]). [20]

1.1.5.6 Used Fuel Reprocessing

Used nuclear fuel can be stored, and eventually disposed of, or it can be reprocessed, taking the usable material out of it to use again. This reprocessing has high dose rates and even higher contamination rates and, as such, is done with extensive use of robots. Different fuel compositions result in different dose rates, however, it is not uncommon to have dose rates as high as $10^3$ Gy/hr during the dismantling process and up to as high as $10^4$ Gy/hr during vitrification of waste materials. [18]

1.1.5.7 Inspection of PWRs

CANDU reactors are not the only reactors that use robotic inspection systems, however, the geometries and amount of shielding present is very different between a
CANDU reactor and a PWR. One notable example of an emergency inspection of a PWR was at the Palo Verde Nuclear Generating Station when a fuel assembly was unable to be removed during a planned refueling. Robotic inspection and repair devices were key in the safe repair and removal of this fuel assembly, especially given that dose rates as high as 5,000 R/hr (43.9 Gy/hr) were noted. [21]

1.1.5.8 Summary

In summary, there are a wide array of robotic systems which operate in radiation environments. These environments all have different difficulties, including radiation field strength. The CANDU inspection environment is unique though in the expected dose rates associated with having fuel so close to the inspection systems are significantly higher than other radiation inspection environments. It is for this reason that it is so important to be able to accurately simulate this environment and ensure that any proposed inspection system will be able to survive.

1.2 Problem Statement

As shown in Section 1.1.4.2, if the coolant pressure needs to be increased for any reason, it will require the fueling machines to be used to form a pressure barrier, as the modified closure plugs are incapable of withstanding operating level pressure. Additionally, the UDM requires the use of the fueling machine bridge, further limiting the possibility for other work to be done at the reactor face at the same time. These limitations limit inspections to being done one at a time. These inspections are very time consuming, taking up to 12 hours per channel, thereby requiring long outages.

Minimizing the length of reactor shutdowns is exceedingly important given the large costs associated with lost revenues due to the reactors not producing electricity. With
this in mind, it is clearly desirable to reduce inspection times in order to reduce reactor
downtime. One method to reduce these inspection times would be to develop a system
that would be capable of inspecting multiple channels simultaneously. Such a system
would need to be self contained and autonomous in its inspections, as well as having
a system of ensuring the pressure boundary is maintained regardless of circumstances.

1.2.1 Improved Inspection System

The design of such an inspection system is being undertaken by a large, interdis-
ciplinary team of researchers at the University of Ontario Institute of Technology’s
Mechatronic and Robotic Systems (MARS) Laboratory. This inspection system will
be capable of performing any inspection that the current CIGAR system can. In
addition, the improved system shall include some form of channel closure that will
be capable of withstanding the full operating pressure of 11 MPa [4], plus a safety
margin. This will ensure that increasing coolant pressure will not require the use of
the fueling machines to maintain the pressure boundary, thereby allowing inspections
of multiple channels to be undertaken simultaneously via the use of multiple copies
of the new inspection system.

The improved inspection system will have the following key features:

- Use the existing CIGAR inspection head
- Be able to perform any inspection the current system can
- Be fully contained within the end fitting once installed
- Include a full pressure closure plug
- Not require the use of the fueling machine bridge once installed
- Include fail safe recovery options
- Allow for multiple copies to simultaneously inspect different channels
This improved inspection system will undoubtedly have significantly more, and more sensitive, electronics within the reactor. This use of more electronics will result in a greater affect of radiation on the components used in the inspection system, and could lead to shortened lifetime of the inspection system. This thesis represents the investigation and analysis of the dose rate and expected doses to the offline pressure tube inspection system that is being developed. In addition, potential radiation effects to critical components of such a system will be considered. These radiation effects, coupled with the dose rate estimations will allow for prediction of potential failure times for the key components within the new inspection system.

Currently there is no data or tool available for mapping or accurately predicting the radiation fields within the reactor as a function of location. This thesis will examine the development of a simulated reactor environment and Graphical User Interface (GUI) to allow for predictions of these dose fields to be done without the need for specially trained personnel.

1.2.2 Objectives

The objectives of this work are as follows.

- To develop a Monte Carlo N-Particle (MCNP) model of the radiation environment within an offline CANDU reactor.
- To estimate the dose rates and total dose that an inspection system is likely to encounter while inspecting an offline CANDU reactor.
- To benchmark the MCNP model against industry values.
- To develop a simple and rapid Graphical User Interface (GUI) that will allow users to generate and run the necessary MCNP simulations, without requiring specialized knowledge of MCNP.
1.3 Scope

The scope of this work is to build an offline MCNP model of the Darlington CANDU reactor. This model will simulate the inside of the reactor channel, however, it will not be able to estimate dose on the reactor face. Additionally, this model will only simulate gamma radiation because, as mentioned in section 1.1.3, the amount of neutron radiation present in the offline state will be very low. The improved inspection system that is being developed as part of the overarching research is not part of this thesis, and while it may be discussed for context, no part of its design is being presented as finished work here.

The original scope of this work included the possibility of online inspections and the comparison of simulated dose rates to measured reactor face dose. However, it was quickly determined that online inspections would not be possible due to the high dose rates. It was also found that the dose rates at the reactor face are almost exclusively due to activation products, and not from fuel dose within the core. It is for these reasons that these objectives were not pursued further.
Chapter 2

Radiation Effects

Radiation comes in many forms and energies, but regardless of the type of radiation or the energy, there is the potential for it to cause damage. Different types of radiation cause damage in different ways, and there are four main categories or radiation: photons, including gamma rays, x-rays, and bremsstrahlung; electrons, such as beta emissions; heavy charged particles, such as alpha particles, protons and other heavy ions; and finally, neutrons. As stated in Section 1.3 this work is only concerned with gamma radiation. This is because the reactor is in a shutdown state, and as such, limited neutrons will be present. In addition, both heavy charged particles and electrons have very short ranges, especially through denser materials, such as water and metals, and as such, none of these particles would be able to reach the area of interest. Given that this work is only considering gamma radiation, this chapter will focus on the effects caused by photons.

There are two main categories when it comes to radiation effects: the effect of radiation on passive components, such as metals, polymers, lubricants and other passive materials; and the radiation effects to active components, such as electronics. Electronics are generally much more sensitive to radiation than passive components are,
and even if the damage occurs when the electronics are not in use, it can still impact the active operation. Each of these categories will be discussed in a separate section below. However, before the effects on materials can be discussed, the mechanism by which radiation interacts with materials must be mentioned.

2.1 Radiation Interaction With Materials

2.1.1 Neutron Interactions

Neutrons are the obvious starting point for this analysis as they are central to nuclear fission. That being said, given that this work focuses on offline reactors where neutrons are not present in large quantities, most neutron interactions will not be covered. Neutrons interact with the nucleus of the atom and can cause a number of changes to the atom. Generally speaking, the neutron is absorbed by the nucleus, following which several different things can occur depending on the atom. The results of the neutron absorption can range from the neutron being ejected at a different angle after depositing some of its energy (called a scatter), to the neutron causing the atom to break into smaller pieces (called fission). In addition, if the neutron stays in the atom after it is absorbed it will cause the atom to become a different isotope. This isotopic change can lead to changes in material properties, as well as possibly changing the atom to a radioactive isotope. These are some of the reasons that inspections are done while the reactor is shut down.

2.1.2 Gamma Interactions

In the offline reactor there are still large amounts of radiation present, however, it is in the form of alpha, beta or gamma radiation. Both alpha and beta radiation have short ranges, especially in denser materials such as metal and water. For this reason the specific interactions of alpha and beta radiation will not be discussed. Photons
such as gamma rays on the other hand, can penetrate deep into materials, even very dense materials. This makes photons the primary threat to materials and electronics used within the reactor.

Photons are classified as ionizing radiation, meaning that they interact with the electrons that surround the nuclei, as opposed to neutrons which interact with the nucleus. This means that when a photon interacts, provided that it has sufficient energy, it will knock electrons out of their orbit, leaving the atom as an ion. This process is called ionization. Photon radiation can interact in one of three ways depending on the energy of the photon, and the attenuation factor of the material, as shown in Figure 2.1. The three interactions that can occur are: photoelectric effect, Compton scattering, and pair production.
2.1.2.1 Photoelectric Effect

Photoelectric effect is when a photon strikes an orbital electron with sufficient energy to knock it from its orbit. The photon is absorbed in this process, transferring all of its energy to the electron. The electron then flies off with energy equal to the incident photon, minus the binding energy of the orbit. This process dominates at lower photon energies, and leads to the creation of free electrons. [23,24]

2.1.2.2 Compton Scattering

Compton scattering is a process that can occur at almost any energy, but is dominant at intermediate energies. Compton scattering occurs when a photon hits an electron and is scattered off at an angle, knocking the electron from its orbit in the process. The angle that the electron is released at is relative to the angle that the photon is scattered at and, additionally, the energy of both the electron and the photon are functions of the angle of scatter. [23,24]

2.1.2.3 Pair Production

Pair production occurs when the photon energy is above $1.022 \text{ MeV}$, and involves the generation of an electron-positron pair. This process is a direct conversion of energy into matter, and has a threshold energy of $1.022 \text{ MeV}$ because this is the rest mass of two electrons ($0.511 \text{ MeV}$). If the energy of the photon is higher than the threshold, the excess energy is split between the electron and positron pair. The electron and positron fly in opposite directions, and after a short distance the positron will come in contact with an electron, resulting in the annihilation of the electron and positron, and the release of two $0.511 \text{ MeV}$ photons in the process. [23,24]
2.1.2.4 Secondary Electrons

The electrons generated from any of these interactions can cause additional ionization damage and will eventually come to rest somewhere else in the material. Ionizing radiation can be especially damaging to molecules that are held together with covalent bonds, as the removal of a bonding electron leads to the breakage of molecular bonds. This breakage causes the disintegration of the existing molecule, and the formation of new molecules, as discussed below. Additionally, ionizing radiation can be very damaging to electronics where both the free electrons, and the ions can cause problems. This will also be examined further later in this chapter. [25, 26]

2.2 Passive Components

Passive components make up a large part of any robot, accounting for any structural component and many other key pieces of the system. Different groups of materials behave differently, and as such, this section has been subdivided into material groups, for ease of understanding.

2.2.1 Metals

The first, and often most common, passive component material that will be discussed are metals. Metals have a fairly unique atomic structure where the atoms are held in a crystal lattice, surrounded by the electrons. These electrons are not attached to any particular atom and can move freely. This free movement means that when a gamma ray ionizes an electron from a metal, another electron can flow in to fill the gap. The displaced electron will come to rest elsewhere in the material, and the steady state of the metal is renewed. This ability of electrons to move freely within metals makes all metals highly resistant to damage caused by ionisation. [25]
2.2.2 Polymers

Polymers, including plastic, are materials comprised of long chains of molecules. These long chains are very susceptible to damage from ionizing radiation, given that these long chains of molecules are what give the polymer its properties. When ionizing radiation interacts with a polymer it causes the bonds of these chains to be broken. Once these bonds are broken several different outcomes may occur, as outlined below. [25]

The shorter chains may stay that way, resulting in a softer and weaker polymer, potentially even causing the polymer to liquefy. Teflon, Plexiglas, and Lucite are all examples of polymers that behave this way. [25]

Alternatively, these small chains could attach to other chains, either at the ends thereby creating longer chains, or in the middle, leading to cross linking. Cross linking is when two separate chains are joined mid-way through the chain by a smaller segment. This results in the polymer becoming more rigid and brittle, and is common in polyethylene, polystyrene, silicon, and neoprene. [25]

Different polymers can tolerate differing amounts of radiation before damage begins to occur. Mylar, for example, can tolerate $3 \times 10^4 \text{ MGy}$ before any significant damage sets in. Teflon on the other had can only tolerate 100 MGy to reach the same level of damage. Figure 2.2 shows the radiation hardness of some common polymers. [25]

2.2.3 Rubbers

Rubbers, more generally refereed to as elastomers, are materials that are made of long chains of atoms, not to be confused with polymers, which are chains of molecules. The properties of elastomers are highly dependant on a balance between the freedom of the chains, allowing for motion and thereby stretch, and the degree of cross linking, giving
Figure 2.2: Effects of Radiation on Various Polymers (Modified from [25])
structural integrity and strength. The degradation mechanisms for rubbers are almost identical to that of polymers, where breakage, and potential cross linking, of the chains can lead to either softer and weaker material, or harder and more brittle material, depending on the composition of the rubber. Butyl rubber, for example, will soften, whereas natural rubber will become hard and brittle. Different rubbers have different radiation hardness, however, natural rubber is more resistant to radiation damage than the synthetic variety. As shown in Figure 2.2 natural rubber can withstand approximately $1 \times 10^3$ MGy before any significant damage occurs. [25]

2.3 Effects on Electronics

Electronics are a major component of any robotic system, and as such, the radiation effects are key to consider. Unlike the passive components discussed in Section 2.2, electronics are active components. This means that they are regularly actively receiving and processing data, often in the form of voltage signals, and this data can often be influenced by radiation interactions, or by the damage left from radiation interactions. This makes electronics potentially much more susceptible to radiation damage than passive components and often makes them the limiting factor in a robotic system lifetime. [19]

The mechanisms by which radiation damages electronics is very complicated and, while important to this work, is not the focus of this project. As such, an overview of radiation effects on electronics has been included, but further details have been left out. For further information on the specifics of these radiation effects to electronics, please see [27–29], or the Nuclear and Space Radiation Effects Conference (NSREC) short course notes, many of which contain a great deal of information on this field.
This section will briefly address the main types of radiation events that are consider-
ered for electronics. Radiation effects in electronics can be divided into one of these
categories: Total Ionizing Dose (TID), neutron fluence effects, and Single Event Phe-
nomena (SEP) [19].

2.3.1 Total Ionizing Dose

As stated in [19], TID is the primary driver for radiation damage to electronics for
most robotic systems. It is a measure of the total amount of ionizing dose that the
electronics have received, regardless of the source. This means that even low energy
beta particles can have a significant effect, provided the total dose is sufficient.
As discussed in Section 2.1, when ionizing radiation, be it photons or secondary elec-
trons, interacts with material, electrons are ejected, producing free electrons and ions.
When this occurs in electronics both the free electrons and the ions, called holes, can
cause damage. Electron-hole pairs are generated along the path of the incident parti-
cle. Some of these electron-hole pairs will recombine, resulting in no damage, however
the fraction of electron-hole pairs that does recombine is a complicated function of
the material, incident radiation, and the electric field that is applied. Experimental
values are used to predict the number of electron-hole pairs that are generated and
survive recombination. For SiO$_2$ it has been found that it takes $\approx 17$ eV to produce
one electron-hole pair. [26]

Following the generation and recombination of these electron-hole pairs, the free elec-
trons are swept away by the electric field, and are collected within picoseconds. The
holes meanwhile, cannot actually move, as they are held in place in a lattice. However
the random movement of the bound electrons fills one hole while generating another
at a different location. Through this random electron movement the holes are able to
“move” without the atoms changing location within the lattice. When the holes arrive
at the SiO$_2$/Si interface a certain percentage are trapped. This trapped percentage can vary from over 50% in commercial devices, to below 5% in specially radiation hardened devices. These trapped holes will anneal over time by combining with excess electrons, thereby neutralizing the damage. \cite{26}

These holes, that are trapped at the SiO$_2$/Si interface, are often referred to as interface traps. The buildup of these interface traps is a time dependant effect because, as mentioned previously, the holes cannot move freely and must instead rely on the random movements of bound electrons to make their way to the interference layer. These traps affect electronics primarily by changing voltages. This is most apparent in Metal-Oxide-Semiconductors (MOS) circuits, due to their reliance on threshold voltages. MOS circuits use a high/low voltage to change the switch from on to off, or vice versa, depending on the type. This means that MOS circuits are susceptible to these traps due to the change in the threshold voltage. Large concentrations of these traps can even lead to the MOS circuit being rendered useless if the voltage threshold shift is large enough to prevent the switch from changing states, thereby leaving it either permanently on or off. Additionally, damage to the electronics can occur at the time of radiation interaction, with the electron-hole pairs causing photocurrents and space charge effects. \cite{26}

### 2.3.2 Other Damage Mechanics

In addition to TID there are two main types of radiation damage that can occur: Neutron damage and SEP. Neutron damage, as the name implies, requires neutrons to occur and SEP require high energy charged particles to occur. Given that neither of these are present in the case scenario being addressed by this model, the descriptions of these events is being included for completeness sake, and will not be overly detailed. \cite{19}
Neutron damage is based on the principles of displacement damage. Due to the neutral charge of the neutrons, they interact with the atoms, unlike gammas which interact with the electrons. This interaction with the atoms held in the crystal lattice can lead to a wide array of damages. Most of these damage mechanisms begin when a neutron hits an atom and deposits enough energy to knock it out of its lattice location. This displacement leaves a vacancy in the lattice and when the atom comes to rest further into the material it can either fill en existing hole, or become an interstitial atom. Lattice vacancies and interstitial atoms disturb the uniform nature of the lattice, and can change a number of properties, which can lead to device failure for electronics. [19]

Displacement damage requires a certain neutron fluence to accumulate before the damage becomes significant. The fluence required varies depending on the damage induced and the electronics that are being damaged, however, fluences of $1 \times 10^{12} \text{n/cm}^2$ to $1 \times 10^{15} \text{n/cm}^2$ are generally required as minimums before damage begins to accumulate [27, 28, 30, 31].

The final damage mechanisms are SEP, which occurs when a single, high energy particle strikes electronics. Generally, these high energy particles are only a threat to space electronics, where fast moving charged particles are present, and as such this will not be a threat to the electronics in the scenario being considered. SEP can be broken down into two main types: strikes that result in *soft* errors, called Single Event Upsets (SEUs); and those that result in *hard* errors, such as latch-up and burnout events. Soft errors are considered those that can be corrected and do not cause permanent damage. Hard errors are those that cause some form of permanent damage to the electronics. [32]
When a high energy charged particle passes through material it loses energy by ionizing atoms along its path. Unlike with photons, which only ionize some of the material, high energy charged particles ionize so much that they cause a dense plasma of electron-hole pairs to be generated along the path. The electron hole pairs act as described in Section 2.3.1 but there are many more of them. Additionally, the plasma itself can cause damage. Soft errors are generally caused by the pulse of electricity generated from the electron hole pairs, and can be dealt with pragmatically. That being said, if these errors occur too frequently it can overwhelm the error handling architecture and still lead to damage. Hard errors on the other hand generally form as a result of the plasma generation. This plasma can cause shortcuts through gates to be generated, and the heat generated from this can be sufficient to melt key components, referred to as burnout. Alternatively, if the burnout does not destroy the device, but instead connects certain components that are designed to be separated, the switch can be rendered useless due to the short circuit caused by the plasma hole. This is referred to as latch-up. There are other damages and effects that SEP can cause, however, these are the primary ones and, as stated earlier, this is not a threat in this scenario.

2.3.3 Radiation Hardness

Given that TID is the primary cause of damage to the electronics in the environment that this work is focused on, it is reasonable to assess the electronics on the grounds of a failure dose. This value, usually referred to as the radiation hardness, represents the dose at which the electronics will cease functioning properly. This does not mean that the electronics will function perfectly and then suddenly stop working, it is actually when the electronics are likely to stop producing reliable results. This means that while the electronics may continue to work past their maximum dose, they are not likely to function well, and will be more likely to have issues. For reliability reasons,
these manufacturer radiation hardness values are often treated as a hard maximum dose, and the electronics are replaced before they reach this dose. Table 2.1 shows the radiation hardness for some components which are likely to be included in the final robotic design. These values for radiation hardness will be used later in Chapter 5 to compare against predicted doses, and will allow for the estimated robot lifetime to be determined.

Table 2.1: Radiation Hardness of Various Robotic Components

<table>
<thead>
<tr>
<th>Type</th>
<th>Component</th>
<th>Total Dose</th>
</tr>
</thead>
<tbody>
<tr>
<td>Computing</td>
<td>Integrated Circuit (CMOS) [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>Microprocessor [33]</td>
<td>3 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>RAM [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>A/D Converter [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>Bus Interface [33]</td>
<td>3 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>SERializer/DESerializer [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td>Location Sensors</td>
<td>Optical Encoder (Standard) [34]</td>
<td>1 kGy</td>
</tr>
<tr>
<td></td>
<td>Optical Encoder (Hardened) [34]</td>
<td>1 MGy</td>
</tr>
<tr>
<td></td>
<td>Optical Encoder (Fibre) [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Resolver [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Potentiometer [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td>Sensors</td>
<td>Ultrasonic Sensor [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Distance Sensor [35]</td>
<td>20 MGy</td>
</tr>
<tr>
<td>Other Components</td>
<td>Conductors/Connectors [35]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Drive Mechanisms [35]</td>
<td>10 MGy</td>
</tr>
</tbody>
</table>
Chapter 3

Monte Carlo N-Particle Simulation

MCNP was selected to be used as the Monte Carlo simulation code that would be used for this work. There are a number of other Monte Carlo simulation codes available, each with their own benefits and specialties. MCNP was selected as the best for this work because it is a well-verified software [36] and is an industry standard in North America.

According to the Monte Carlo Team at Los Alamos National Laboratory (LANL) [37], MCNP “is a general-purpose three-dimensional simulation tool that transports 37 different particle types for criticality, shielding, dosimetry, detector response, and many other applications.” To put this another way, MCNP is a radiation transport code that uses the Monte Carlo method to simulate the transportation of multiple types of radiation, at different energies, through a material. This chapter will explain the basics of the Monte Carlo method, as well as some specifics on MCNP before explaining the model that was developed.
3.1 Monte Carlo Method

The Monte Carlo method is a stochastic modeling method that uses random numbers to simulate real world scenarios. Monte Carlo methods are often used to simulate the transportation of radiation through a medium, given the random nature with which radiation behaves. In this case, the Monte Carlo method consists of simulating a finite number of particle tracks, known as histories, through the use of pseudo-random numbers, as it is transported through a material.

In each history, random numbers are generated and sampled from Probability Density Functions (PDFs) in order to simulate the movement of the particle. The random numbers are used to generate many different traits of the particle including: initial position, initial energy, direction of travel, track length, etc. These traits can be treated as random variables which take on certain values with a frequency determined by the PDF. The PDF represents the likelihood that the random variable, \( x \), will take a value between two bounds, \( a \) and \( b \), as [38]:

\[
P\{a \leq x \leq b\} = \int_{a}^{b} f(x)dx
\]  

(3.1)

The PDF will need to be normalized, depending on the bounds of the random variable. If the variable can take on any real value between \(-\infty\) and \(\infty\) then it will be normalized as [38]:

\[
\int_{-\infty}^{\infty} f(x)dx = 1
\]

(3.2)

Alternatively, if the random variable has a restricted domain then the PDF will be normalized as [38]:

\[
\int_{x^-}^{x^+} f(x)dx = 1
\]

(3.3)
In addition to the PDF, there is also the cumulative probability distribution function. The cumulative probability distribution [38]:

\[ F(x) = P\{x' \leq x\} \quad (3.4) \]

is the probability that a random variable \( x' \) will take on a value equal to, or less than \( x \). Given the definition of \( f(x) \), it is known that the random variable can only take on real values.

This limitation, coupled with the definition of the probability distribution shown in Equation (3.4), allows the cumulative probability density to be defined as [38]:

\[ F(x) = \int_{-\infty}^{x} f(x')dx' \quad (3.5) \]

It can clearly be seen from Equation (3.5) that the following are also true [38]:

\[ \lim_{x \to \infty} \equiv F(\infty) = 1 \quad (3.6) \]

\[ \lim_{x \to -\infty} \equiv F(-\infty) = 0 \quad (3.7) \]

As noted earlier, Monte Carlo calculations rely on random numbers. True random numbers are not achievable in computer calculations and, as such, pseudo-random numbers are used. These pseudo-random numbers, \( \xi \), are uniformly distributed such that [38]:

\[ 0 < \xi \leq 1 \quad (3.8) \]
Given this distribution, $\xi$ can be used to sample $F(x)$ by [38]:

$$F(x) = \xi$$

(3.9)

This gives an unbiased sample of $F(x)$, however, it is the distribution of $x$ that is of interest. This can be found by performing an inversion of the function after each random number is generated. This allows $x$ to be sampled by [38]:

$$x = F(\xi)^{-1}$$

(3.10)

By performing this inversion any random variable that has a cumulative probability distribution defining the physical phenomenon can be sampled. This process is key to the Monte Carlo method but can be difficult to perform depending on the complexity of the PDF and the cumulative probability density. There are three methods by which this inversion can be done: direct inversion, acceptance/rejection method, and discrete data. [38]

### 3.1.1 Direct Inversion

Direct inversion is the ideal method to sample $x$ but is often too complex to be feasible, if it is even possible. There are, however, some simple inversions, such as the distance between collisions, that can be directly inverted. Direct inversion will be demonstrated here using the example of calculating the distance a particle travels between collisions. [38]

To calculate the distance between collisions, measured in mean free paths, as described by $0 \leq x \leq \infty$, the PDF is [38]:

$$f(x) = e^{-\mu x}$$

(3.11)
and the cumulative probability density is [38]:

$$F(x) = \frac{1 - e^{-\mu x}}{\mu} \quad (3.12)$$

These equations are a mathematical representation of the physical phenomenon. The PDF represents the probability that a collision will occur within the range of $x$ to $x + dx$, and the cumulative density function represents the probability that a collision will occur within distance $x$. Due to the simplicity of this equation, a direct inversion can be done as [38]:

$$x = -\frac{\ln(1 - u\xi)}{\mu} \quad (3.13)$$

Using this inverted equation, $\xi$ can be subbed into Equation (3.13) and $x$ can be directly calculated.

#### 3.1.2 Acceptance/Rejection

The direct inversion is the ideal inversion method, however, there are many times where this is too complicated. When this is the case, it is often beneficial to use the acceptance/rejection method. For any random function it can be defined that $0 \leq x \leq a$ and that $0 \leq f(x) \leq f_{\text{max}}$. Using this, $x'$ and $f(x')$ can be defined, such that when graphed they fit within a unit square. [38] These are defined as [38]:

$$x' = \frac{x}{a} \quad (3.14)$$

$$\tilde{f}(x') = \frac{f(ax')}{f_{\text{max}}} \quad (3.15)$$

In order to sample the distribution of $x'$, and by extension the distribution of $x$, two random numbers, $\xi_1$ and $\xi_2$, are generated, evenly distributed between 0 and 1. This
allows for $x'$ to be set as $x' = \xi_1$ and then check if $\xi_2 \leq \tilde{f}(\xi_1)$. If this is found to be true then the pair is accepted, otherwise it is rejected. As shown on Figure 3.1, this can be visualized as checking if the point $(\xi_1, \xi_2)$ falls underneath the curve formed by $\tilde{f}(x')$. If it is under the curve it is accepted, otherwise it is rejected. [38]

This method is very useful for cases where $F(\xi)^{-1}$ is hard to find because the inversion can be completed without finding this. However, this method is ineffective when the area under the curve $\tilde{f}(x')$ is small compared to unity. This is because most random number pairs will fall outside of the curve and will be rejected. [38]

### 3.1.3 Discrete Data

In Monte Carlo methods, it is common that PDFs are given in the form of numerical data from a histogram, which represents physically measured values, as seen in Figure 3.2a. As can be seen from Figure 3.2b, the cumulative probability distribution is a piece-wise, linearly increasing function, and can be described by [38]:

$$F(x) = \frac{1}{x_i - x_{i-1}}[(x - x_{i-1})F_i + (x_i - x)f_{i-1}]$$  (3.16)
Figure 3.2: (a) Probability Density Function and (b) Cumulative Probability Distribution for Discrete Data

where

\[ F_i = \sum_{i'}^i f_{i'}(x_{i'} - x_{i'-1}) \]  

(3.17)

From here it is possible to indirectly determine \( x = F(\xi)^{-1} \). First, the interval of the cumulative probability density function that \( \xi \) is in is determined, such that \( F_{i-1} \leq \xi \leq F_i \). From this, \( F(x) \) is set equal to \( \xi \) in Equation (3.16), and then is solved as [38]:

\[ x = \frac{(x_i - x_{i-1})\xi - x_iF_{i-1} + x_{i-1}F_i}{F_i - F_{i-1}} \]  

(3.18)

Discrete sampling is also used for times when a value of \( x \) is not needed, but instead, an answer to a question is needed, such as “What type of collision has occurred?” This is done by dividing the cross section (\( \sigma \)) for each individual interaction by the total cross section. These cross sections represent the probability that the specific interaction occurs. This means that the total range of all of the fractions together is from 0 to 1. Using this, if each ratio is set to have a specific range, a random number \( \xi \) can be generated and compared to the ranges for the interaction types. Whichever range \( \xi \) falls within is the interaction that occurs. [38]
3.2 MCNP Codes

MCNP6.1 [37] is the most recent version of MCNP, and is what has been used for this work. Before MCNP6.1 there were two separate MCNP codes, each with unique functionality [37]. MCNP5 [39] was the core MCNP code and is able to deal with a wide energy range of neutrons, photons, and electrons. In addition to shielding and dosimetry type calculations, MCNP5 is also capable of performing criticality calculations, including the capability of determine the $k_{eff}$ eigenvalues.

Monte Carlo N-Particle eXtended (MCNPX) [40] was designed to offer several extended options that are not available in MCNP5. These extensions include, but are not limited to, the ability to simulate the transportation of over 30 different particles, extended energy ranges, and several improvements to the physics models.

MCNP6.1 [37] represents a combination of the core MCNP5 with the extended functionality of MCNPX. MCNP6.1 is more than just a merger however, and includes a number of new developments and improvements. MCNP6.1 is now capable of simulating 37 different particles at a wide range of energies. These energy ranges, which are particle dependant, are shown in Figure 3.3, along with some of the proposed extensions. The area of interest for this work has been circled for clarity.

3.2.1 MCNP Input

Regardless of which MCNP code is used the basic input structure is very similar. Users provide an input file, referred to as an input deck, that must contain certain cards, or lines of code, in a specific order. This naming convention of decks and cards goes back to early computing, when MCNP, which is built on a FORTRAN engine, used punch cards as input. [39]
MCNP is capable of simulating almost any situation, however, it requires the user to provide an input deck to describe the specific situation to simulate. Each MCNP input deck must contain specific blocks of data, and must appear in a very specific order, as follows [39]:

**Title Card**  The title of the project, will be ignored by MCNP.

**Cell Cards**  Uses the surface cards and Boolean math to build cells.

  <Must be separated with a blank line>

**Surface Cards**  Define surfaces using basic shapes (i.e. sphere, planes, etc.) or by using macro-bodies (a pre-made shape, such as a box).

  <Must be separated with a blank line>

**Data Cards**  Contains information such as source declaration, tally declaration, materials, number of particles, and more.

  <Recommended to add a blank line>
It is key that the input is entered in the correct format, and without the use of control
characters, such a “tabs”, because a failure to do so will result in the simulation
crashing. In addition, aside from the blank lines to separate the different sections,
there cannot be any other blank lines. Once the input deck has been created, the
code is called from a command prompt and the input files are specified. Output file
names can be specified, otherwise default names are used. [39]

3.2.2 Verification Cases

Before the model of the reactor was developed several simplified cases were run using
MCNP to ensure that the results were acceptable. Several simple shielding arrange-
ments were set up and simulated in MCNP and MicroShield, a deterministic, point
kernel modeling software, as well as being compared to hand calculations. The test
cases included comparing calculation of gamma constants, simple shielding arrange-
ments, and a simplified CANDU fuel channel. All of these tests came back with
matching results thereby showing that MCNP was able to accurately model particles
in the energy ranges of interest.

3.3 Model

As discussed in Section 1.3, the reactor that is being modeled in MCNP for this sim-
ulation is the Darlington CANDU reactor. The Darlington reactor is a 480 channel
CANDU reactor. This section will discuss the major components within the MCNP
model and explain their interactions. Figure 3.4 shows a 3D rendering of the com-
pleted reactor model. In order to build this model, many dimensions were needed.
Many of these dimensions came from Ontario Power Generation (OPG) in the form of
technical drawings [41]. Please see Appendix A for a table outlining the drawings used.
Figure 3.4: Full Reactor Model
This section will describe each of the individual components of the model, as well as how they interact with each other. Throughout the section will be a number of figures extracted from VISED [42], a visual editor for MCNP, which has been used to confirm geometries and to produce graphic representations of the model. These figures are automatically coloured according to material, and the colours will be used to reference specific components within the figures.

3.3.1 Fuel Bundle

The first component of the model that will be discussed is the fuel bundle. The fuel bundle used in the Darlington reactor, and thus in the model, is a 37 element bundle. This bundle, as discussed in Section 1.1.1.1, consists of 37 individual elements, or fuel pencils, held together with two end plates. Each fuel element would consist of a sheath, made from zirconium alloy, and a number of fuel pellets. For this model the pellets have been simplified to one continuous piece, and the end plates, which hold the elements in the right configuration have been ignored. Neither of these changes should make significant difference as the amount of material present has not changed significantly, nor has the configuration of shielding changed.

Figure 3.5 shows a 3D rendering of the fuel bundle as it appears in the MCNP model. In addition to the fuel and sheathing, the D$_2$O coolant volume around the fuel is also modeled. The coolant is set to take up any location not already occupied by the fuel bundle, and is left intentionally larger than needed in order to be cropped to the appropriate size by the containing assembly.

3.3.2 Fuel Channel Assembly

The next component is the fuel channel assembly, composed of the fuel string, pressure tube, annulus gap, calandria tube, and the surrounding moderator. The fuel
bundle, which was discussed in the previous section, is put into an lattice, and repeated a total of 13 times. This fuel string lattice is bounded by the Zircalloy-2.5Nb pressure tube, which cuts the lattice to the proper dimensions, and ignores anything outside of the pressure tube inner surface. Around the pressure tube is the Zircalloy-2 calandria tube, with a CO\textsubscript{2} annulus gas filling the space between. A large area of D\textsubscript{2}O moderator is then placed around the calandria tube. This excess moderator, like the excess coolant around the fuel, will later be bounded by the lattice parameters.

Figure 3.6 shows a cut through of this assembly, with all of the individual components visible. The fuel elements can be seen in the centre (dark blue) with the fuel sheath (barely visible) and coolant (green) around it. Around that is the pressure tube (light blue), the CO\textsubscript{2} annulus gas (yellow) and the calandria tube (blue). Some of the moderator (green) can also be seen, however, the moderator extends much further.
3.3.3 Side Structural Components

At either end of the fuel channel assembly is a number of Side Structural Components (SSCs), including the end fittings, closure plugs, shield plugs, and shield walls. The following subsections will describe each of these components separately.

3.3.3.1 End Fitting

The first of these SSCs that will be discussed is the end fitting. The end fitting is a large structure, which attaches to the fuel channel assembly at either end. The end fitting is where the feeder pipes attach, allowing for the coolant to be circulated. Additionally, the end fitting allows for the fueling machine to attach and form a pressure boundary, thereby allowing online refueling, by removing the closure plug from the end fitting. In addition to the closure plug, the end fitting also houses the shield plug. This shield plug will be discussed further in the next section. The final component in the end fitting assembly is the liner tube, which provides a channel for the incoming coolant to flow through, and allows the fuel bundles to pass through the end fitting and into the pressure tube.
Figure 3.7: Section View of the End Fitting

Figure 3.7 shows the end fitting, as well as two section views of important parts. In this figure, the end fitting can be seen (yellow), along with the shield plug. The closure plug can also be seen at the end of the end fitting, however, it is worth noting that for simplicity sake, the closure plug has not been modeled as a separate component, but instead as a part of the end fitting. The first section view shows where the pressure tube (blue) joins into the end fitting. This joint is the rolled joint as discussed in Section 1.1.1.1. The second section view shows the liner tube, which is difficult to see in the full view. It is also worth noting that while some simplifying assumptions have been made to the dimensions and shape of the end fitting assembly in order to allow for modeling, these simplifications preserved the thickness and the total volume of the material and will not have an affect on the dose.
3.3.3.2 Shield Plug

The shield plug is a large piece of stainless steel which acts as a biological shield, in conjunction with the shield wall. The shield plug is located within the end fitting and is held in place by a set of locking jaws which interact with corresponding mechanisms on the inner surface of the end fitting. While the shield plug does provide shielding, it also serves two other functions.

The first of these other functions is to hold the fuel string in place. With one shield plug on either side, the space between the plugs is set such that it holds the fuel string in place, thereby preventing it from moving axially. This is important because any small movement of the fuel can lead to significant changes to the overall reactor power distribution, and can even lead to transients.

The second function of the shield plug is to direct and smooth the flow of the coolant as it enters the channel. As mentioned in the last section, coolant enters the end fitting and travels along the annulus between the end fitting and the liner tube. Near the end of this annulus the liner tube has a number of holes in it, which lines up with a set of openings, or flow channels, in the shield plug. Figure 3.8 shows a 3D rendering of the shield plug and these flow channels can clearly be seen. These flow channels lead to the fuel channel and function to direct and smooth the bulk of the coolant flow. Figure 3.9 shows several section views of the shield plug in order to better illustrate these flow holes. Note that there are several different designs for the flow smoothing portion of the shield plug, as well as small differences between the inlet and outlet shield plugs, but for simplicity sake, a single, somewhat simplified, shield plug was used in this model.
Figure 3.8: 3D Rendering of the Shield Plug

Figure 3.9: Section View of the Shield Plug
3.3.3.3 Shield Wall

The shield wall is the primary biological shield that prevents radiation of any kind from coming out of the core. This is especially important when the reactor is in shutdown and work is being done at the reactor face. The shield wall has four main components: the Calandria Side Tube Sheet (CSTS), the Fueling machine Side Tube Sheet (FSTS), lattice tubes, and the shield fill. The CSTS and FSTS are thick slabs of stainless steel, approximately 5.1 cm, and 7.6 cm thick, respectively. These tube sheets have a number of holes in them, one for each channel, and the lattice tubes stretch between the two tube sheets, creating a void between them. This void is filled with carbon steel shot balls and light water, in a 60:40 ratio [43], which acts as the primary shield against both neutron and gamma radiation. The tube sheets and liner tube also form the ends of the annulus gas system. The CSTS has a rolled joint attaching the calandria tube to it, creating an enclosed volume for the moderator on one side, and the annulus on the other. This annulus continues through the gap between the end fitting and liner tube.

Figure 3.10 shows the major components of the shield wall in a section view of an empty fuel channel. The two tube sheets (yellow) can clearly be seen at either end, both above and below the end fitting (yellow). The liner tube (yellow) is more difficult to see, due to its thickness, but it can be seen between the two tube sheets. The carbon steel shot and light water has been homogenized into a single uniform material, with the appropriate properties, and can be seen between the tube sheets (orange). Figure 3.11 shows how the shield wall fits in with multiple channels. It is also worth noting that the shield plug, inside of the end fitting, completes the shield. In the 13 bundle arrangement, which Darlington uses, half of each end bundle protrudes part way into the end fitting, and consequently into the shield wall, as seen in Figure 3.11.
Figure 3.10: Shield Wall Single Channel Close-Up

Figure 3.11: Shield Wall with Multiple Channels
3.3.4 Lattice

The previous sections have described the fuel string, fuel channel, and various SSCs, however, this only forms one of the 480 channels in the reactor. In order to create the full reactor, there are two options: individually define all of the channels and components, or use a repeated structures lattice. Defining all of the channels would be impossible due to the number of surfaces required and the limited number of surfaces able to be used. It is for this reason that MCNP has a repeated structures lattice function.

A single complete channel is modeled and then placed into a lattice cell, and repeated a number of times. The complete channel is made up of the fuel channel assembly, with the fuel string inside, an end fitting on either end of that, with the shield plug inside, and a section of shield wall that extends beyond the lattice cell. The shield wall, moderator, and air around the end fitting all extend well beyond the size of the lattice cell. This is done so that the lattice cell will cut these materials to size, guaranteeing that all locations have the proper material properties. A lattice cell is set up using the lattice pitch of the CANDU reactor, 28.6 cm [4], as the y and z dimensions, and the entire length of the assembled components as the x dimension. This allows multiple channels to be placed side by side, and ensure that the space between channels is correct. Using this lattice cell as a basis, a repeated structures lattice is set up and the single lattice is repeated 479 times. Figure 3.12 shows a 3 by 3 section of the lattice that was made.

MCNP requires a completely filled square lattice for these structures, however the CANDU reactor has a circular shape. To deal with this issue, a separate lattice cell was defined which consisted of moderator in the centre, with a shield wall section on either side, and air beyond that. What this did was extend the existing geometries that
continue beyond the outermost channel, but without adding more channels. These separate lattice cells extend the geometries well beyond what they need to because they will be trimmed to size by being bounded by the outer components, in the same way the coolant around the fuel was bounded.

3.3.5 Outer Components

The last components in the reactor model are the outer reactor components, including: the reflector volume, calandria tank, and several bounding surfaces. The first component is a bounding surface that cuts the excess empty cells in the lattice, as discussed in the previous section, to the proper shape and size. Around this surface is the reflector volume, a large volume of heavy water moderator with no channels, that acts as a shield and to reflect neutrons back into the core during operation. Around that is the calandria tank, which holds the moderator. In addition to these components around the core section of the reactor, there is also additional extensions to the
Figure 3.13: Full Reactor Core Section View

Figure 3.14: Full Reactor Side View
shield wall, allowing it to meet up with the calandria tank, thereby, forming a closed volume. Finally, the air around the end fittings was extended to match the calandria tank, giving one large cylinder for the entire model. Figures 3.13 and 3.14 show the full reactor geometry including all of the internal components.

MCNP requires that there be an ultimate end in every direction. This is accomplished by setting everything outside of the cylinder that makes up the model to be a complete vacuum, called void, and setting the importance of any particle which enters this space to zero. An importance of zero on a particle tells MCNP to ignore it from then on, and thereby deletes the particle. In this way it can be set that any particle which happens to reach the outside of the reactor is deleted, given that they no longer affect the simulation.

### 3.3.6 Source Term

The source term is arguably the most important part of the reactor model, as it is what describes the radiation that is being simulated. In an offline reactor there are two main sources of radiation: the fuel and the activated reactor components. A source term describing the activated reactor components was developed and tested. However, due to a lack of input data, the results produced could not be made to accurately match benchmark data [44]. Given this, an alternative method of adding the activation dose was selected, which will be discussed in this section. The simulation of the activated components was moved to a future work goal. Appendix B discusses the activated reactor components model that was developed, as well as the results produced.

As discussed in Section 1.1.1.1, the fuel in the CANDU reactor is at varying levels of burnup, due to the online refueling. This non-homogeneous fuel profile makes the
development of an accurate source term quite difficult. There are several different approaches that could be used to simplify this problem, each with different degrees of difficulty. The approach that was used in this model is to treat all of the fuel in the reactor as discharge fuel. Discharge fuel has the highest concentration of fission products due to the high burnup. Given that many of these fission products are gamma emitters, discharge fuel has the highest gamma flux, and thereby create the highest dose. Treating all of the fuel as discharge fuel will lead to a higher total dose, as compared to a mixed burnup core, and as such is a conservative estimate of the dose that will be encountered.

The source term that was used to describe the discharge fuel was supplied by Duncan Barber. The source term was supplied in the form of a list of isotopes and corresponding activities, as calculated by an ORIGIN-S [45] simulation as part of Duncan Barber’s Ph.D. thesis [46, 47]. In addition to the individual isotopes and activities, the ORIGIN-S output includes a table of gamma energy release rates. These gamma energy release rates are a combined representation of all of the gamma emissions, accounting for different emission probabilities, from all of the present isotopes. Given that MCNP does not deal with individual isotopes, but instead deals with gamma emissions, this output from ORIGIN-S can be directly input into MCNP. Furthermore, the ORIGIN-S simulation included the fuel at nine different decay times, ranging from 0.2 hours after shutdown to 240 hours after shutdown. These different decay times will allow more versatility in the MCNP simulation, as they can be used to accurately simulate how long after shutdown the inspections began. Table 3.1 shows the gamma energy emission rates for several key decay times. Most of the simulations were done using the 7 day decayed fuel term, as this is commonly when inspections are started [48], however other runs were done for comparison.
### Table 3.1: Gamma Energy Emission Rates at Several Key Decay Times [46]

<table>
<thead>
<tr>
<th>Energy Bounds (MeV)</th>
<th>Emission Rate (γ/s)</th>
<th>1 day</th>
<th>3 day</th>
<th>7 days</th>
<th>10 days</th>
</tr>
</thead>
<tbody>
<tr>
<td>0 – 0.015</td>
<td>1.84E+14</td>
<td>1.08E+14</td>
<td>4.55E+13</td>
<td>2.82E+13</td>
<td></td>
</tr>
<tr>
<td>0.015 – 0.02</td>
<td>2.06E+13</td>
<td>1.39E+13</td>
<td>9.44E+12</td>
<td>7.96E+12</td>
<td></td>
</tr>
<tr>
<td>0.02 – 0.03</td>
<td>5.69E+13</td>
<td>3.83E+13</td>
<td>2.38E+13</td>
<td>1.87E+13</td>
<td></td>
</tr>
<tr>
<td>0.03 – 0.04</td>
<td>9.26E+13</td>
<td>6.66E+13</td>
<td>4.35E+13</td>
<td>3.44E+13</td>
<td></td>
</tr>
<tr>
<td>0.04 – 0.05</td>
<td>4.68E+13</td>
<td>2.97E+13</td>
<td>1.82E+13</td>
<td>1.47E+13</td>
<td></td>
</tr>
<tr>
<td>0.05 – 0.06</td>
<td>3.15E+13</td>
<td>1.99E+13</td>
<td>1.19E+13</td>
<td>9.47E+12</td>
<td></td>
</tr>
<tr>
<td>0.06 – 0.08</td>
<td>5.63E+13</td>
<td>3.59E+13</td>
<td>2.16E+13</td>
<td>1.73E+13</td>
<td></td>
</tr>
<tr>
<td>0.08 – 0.1</td>
<td>2.46E+14</td>
<td>1.61E+14</td>
<td>7.97E+13</td>
<td>5.25E+13</td>
<td></td>
</tr>
<tr>
<td>0.1 – 0.15</td>
<td>1.74E+15</td>
<td>1.02E+15</td>
<td>4.02E+14</td>
<td>2.33E+14</td>
<td></td>
</tr>
<tr>
<td>0.15 – 0.2</td>
<td>8.99E+13</td>
<td>6.06E+13</td>
<td>4.12E+13</td>
<td>3.45E+13</td>
<td></td>
</tr>
<tr>
<td>0.2 – 0.3</td>
<td>2.05E+15</td>
<td>1.11E+15</td>
<td>3.79E+14</td>
<td>1.84E+14</td>
<td></td>
</tr>
<tr>
<td>0.3 – 0.4</td>
<td>7.84E+14</td>
<td>5.53E+14</td>
<td>3.37E+14</td>
<td>2.54E+14</td>
<td></td>
</tr>
<tr>
<td>0.4 – 0.5</td>
<td>8.41E+14</td>
<td>7.59E+14</td>
<td>6.50E+14</td>
<td>5.83E+14</td>
<td></td>
</tr>
<tr>
<td>0.5 – 0.6</td>
<td>1.28E+15</td>
<td>7.84E+14</td>
<td>5.80E+14</td>
<td>5.14E+14</td>
<td></td>
</tr>
<tr>
<td>0.6 – 0.8</td>
<td>5.32E+15</td>
<td>3.54E+15</td>
<td>2.59E+15</td>
<td>2.30E+15</td>
<td></td>
</tr>
<tr>
<td>0.8 – 1.0</td>
<td>1.03E+15</td>
<td>7.70E+14</td>
<td>5.36E+14</td>
<td>4.30E+14</td>
<td></td>
</tr>
<tr>
<td>1.0 – 1.5</td>
<td>8.08E+14</td>
<td>3.73E+14</td>
<td>1.92E+14</td>
<td>1.34E+14</td>
<td></td>
</tr>
<tr>
<td>1.5 – 2.0</td>
<td>2.57E+15</td>
<td>2.30E+15</td>
<td>1.88E+15</td>
<td>1.61E+15</td>
<td></td>
</tr>
<tr>
<td>2.0 – 3.0</td>
<td>2.77E+14</td>
<td>2.33E+14</td>
<td>1.84E+14</td>
<td>1.56E+14</td>
<td></td>
</tr>
<tr>
<td>3.0 – 10</td>
<td>8.72E+11</td>
<td>7.61E+11</td>
<td>6.31E+11</td>
<td>5.41E+11</td>
<td></td>
</tr>
<tr>
<td><strong>Totals</strong></td>
<td><strong>1.75E+16</strong></td>
<td><strong>1.20E+16</strong></td>
<td><strong>8.03E+15</strong></td>
<td><strong>6.61E+15</strong></td>
<td></td>
</tr>
</tbody>
</table>
As mentioned, these energies and emission rates can be used as an input into the MCNP simulation. MCNP takes the energy distribution as an input, but requires a PDF to describe the probability of each individual energy being emitted. The emission rate represents this probability, however, it needs to be normalized to 1. MCNP can do this automatically, but it is preferable to do this in advance in order to ensure that all probabilities are calculated properly. The normalization is done by dividing each emission rate by the total emission rate of all energies, resulting in a fraction between 0 and 1 for each energy range.

In addition to the energies and probabilities of emission, MCNP also requires locations to generate the particles at. Due to the use of the repeated structure lattices in this model, the description of source location must use these repeated structures. Several functions were developed which describe the repeated structure path to each individual fuel bundle. Each of these paths is given a probability that makes particle generation from all locations equally probable. From this, each fuel bundle was described using power laws to evenly represent the entire volume. Power laws were used in order to evenly distribute the probability of particle generation throughout the volume. The fuel bundles used a cylindrical volume which encompassed all of the fuel pencils. The probability of a particle being generated at a given radius is proportional to the volume of a thin slice at radius \( r \), as defined by [37]:

\[
\frac{dV}{dr} = \frac{d(\pi r^2 h)}{dr} = 2\pi rh \quad (3.19)
\]

This shows that the probability of radius \( P(r) \) being selected is proportional to \( r \) and a first order power law can be used. This will distribute the probability of a particle being generated evenly throughout the cylinder. Similarly the axial location can be defined using a zeroth order power law. These two power laws evenly distribute particle generation throughout the cylinder, and from there any particles that do not
fall within a fuel pencil can be rejected, thereby evenly distributing particle generation through the complex shape of the fuel bundle.

### 3.3.6.1 Activation Term

As mentioned earlier in this section, a source term describing the activation products was developed, but due to lack of accurate data, could not be made to reliably match benchmark data. While a fully functional activation term would be desirable, it was determined that there were too many independent variables that could affect such a calculation, hence introducing error. As such, an alternate method for modeling the dose from activation products is needed, and the development of a full activation term has been considered for future work.

The method that was chosen for modeling the activation dose was to estimate a multiplication factor for the fuel dose. Activation data was taken from known sources, and the dose from activation was taken as a ratio to the total dose (activation and fuel). From this factor it can be estimated what percentage of the total dose is due to the activation product, and subsequently, what percent the fuel dose should be increased by to yield an estimate in the total dose. In this way a total dose, including the activation component, can be estimated without the need to actually simulate the activation product.

In order to calculate this dose fraction, a source was needed that included both the fuel only dose, and the activation product dose. According to [44] the dose from the fuel only, in the centre of an empty channel should be

\[
\dot{D}_{\text{fuel}} = 4.91 \text{kGy[Air]/hr} \quad (3.20)
\]
Additionally, it is stated that the dose rate from activated components is

\[ \dot{D}_{activation} = 1.58 \text{kGy[Air]/hr} \]  

(3.21)

This activation product dose is based on a measured value from a fully defueled reactor, six months after shutdown. While a six month shutdown term is not the same as the scenario being simulated, this activation term is an approximation that should provide an indication of the total dose including activation products. Using these dose rates, the activation ratio can be found by finding the total dose rate, and then finding the percentage of that which is from activation products, as shown by

\[ \dot{D}_{total} = \dot{D}_{fuel} + \dot{D}_{activation} = 6.49 \text{kGy[Air]/hr} \]  

(3.22)

\[ \%_{Activation} = \frac{\dot{D}_{activation}}{\dot{D}_{total}} = 24.32 \% \approx 25 \% \]  

(3.23)

It is important to note that this method of estimating dose from activation products has some limitations. Given that the activation dose is directly calculated from the fuel dose using this approach, any locations where there is little to no fuel will result in a proportionally low dose from the activation products. This means that if a section of the reactor were to be emptied of fuel the model would predict almost no dose, where in reality the activation dose would still be significant. This also affects places with large amounts of shielding, such as the end fittings. While these locations should produce some dose from activation products, if there is low dose rates from the fuel, the model will predict low activation dose as well. This problem will be addressed in future work by implementing a more accurate and realistic activation term.
3.3.7 Tallies

With the fuel defined, the entire model has been finished, however, that does not give any way of gathering data from the model. MCNP deals with this through the use of tallies. Tallies are effectively counts of how many of a specific event have happened. Many tallies are some form of a flux tally, i.e., a tally that counts the flux through a specific surface or volume. For this work f4 volume flux tallies have been exclusively used. These tallies work by measuring the track length through the volume of interest, and using this to calculate the tally score, the value added to the tally total, as defined by:

\[ TallyScore = \frac{WT_L}{V} \]  

(3.24)

where \( W \) is the track weight, \( T_L \) is the track length, and \( V \) is the target volume. The track length is the distance that the radiation traveled inside of the target volume before undergoing some sort of interaction or event. A single particle may pass through the target volume multiple times, depending on collisions, and may even have a collision inside of the target volume, thereby increasing the track length.

From this tally score the tally response can be defined as:

\[ f4 = \frac{1}{N} \sum \frac{WT_L}{V} \]  

(3.25)

where \( N \) is the number of source particles simulated. MCNP returns all tally responses relative to the number of source particles (unit/\( S_p \)). This means that all results are independent of the number of particles simulated. Given this, increasing the number of particles simulated will not change the solution being calculated, but will improve the statistics, and therefore the accuracy.
This normalization to the number of source particles means that tallies output results in units of $\gamma/cm^2 S_p$. In order to convert these tallies into a usable flux ($\gamma/cm^2 s$) the results must be re-normalized to the fuel emission rate. In order to do this the tally output is multiplied by the total number of particles released per second by all the fuel in the reactor ($S_p/s$), thereby resulting in a flux ($\gamma/cm^2 s$).

The tallies are all binned based on energy, meaning that when a result is measured it is added to a specific bin based on the energy of the particle that made the track. This binning allows for the flux measurement to be converted into air kerma, through the use of a Dose Conversion Factor (DCF). This DCF, from International Commission on Radiological Protection (ICRP) 119 [49], converts flux, in units of $\gamma/cm^2 s$, into kerma, with a unit of pGy[Air]/s. This is done by multiplying each energy bin by a conversion factor with units of $pGy[Air] cm^2$. These individual energy binned dose rates are also summed up automatically to give a total dose rate for the tally. One final conversion is performed in order to be able to present the results in units of Gy[Air]/hr.

During inspection the fuel channel of interest would be emptied, removing the shield plugs and all fuel. This gives a clear path through the entire channel, allowing for a complete inspection. This same situation was programmed into the simulation through the use of the repeated structure lattice. An empty channel was programmed and can be placed at any location where a channel is. Following this, the fuel source term is updated to prevent source particles being generated in this empty channel. Once the channel is set up tallies can be prepared. Tally volumes were set up throughout the length of the channel, each volume being 50 cm long. This allows for the dose rates to be calculated at multiple locations axially, thereby giving a dose profile.
The primary tally arrangement that was prepared was a non-shielded case. This setup uses a large tally volume, almost filling the entire diameter of the pressure tube, and as such, the tally will statistically converge faster than smaller tallies. Figure 3.15 shows the layout of this tally and the fuel channel can be seen, as outlined in Section 3.3.2 with the moderator around it. Inside of this pressure tube several black outlines can be seen within the heavy water (green). These outlines show the borders of the tally volume, which is 50 cm in length and 10 cm in diameter.

3.3.7.1 Shielding Cases

In addition to being able to simulate the emptied channel the model needs to be able to simulate the improved inspection system, including any shielding it has. This is especially important in order to determine the amount of dose that the sensitive electronics will receive. The inspection system will include a shielded compartment to house the most radiation sensitive components. In order to help determine potential candidate materials, and the thicknesses required, this model must be able to test different materials. With this in mind, a number of tally arrangements were set up, with different thicknesses of shielding. Shielding thicknesses have been set up from 0 cm to 3.0 cm in order to find a sufficient thickness. 3.0 cm was chosen as the maximum thickness because any more shielding than this would not leave sufficient space for the electronics. This variable shield thickness can be simulated as any material, however, for this thesis three candidate materials for the inspection system have been used. These materials are: steel (SS304L), tungsten, and depleted uranium.

The shielding cases all use the same tally volume in order to ensure comparable results. This tally volume is a cylinder that is 50 cm long, and 3 cm in diameter, and uses air as the tally material. This tally volume also has air around it, up to the shield. This is done to keep consistency between the different shielding cases, because the shield will
likely be filled with air. Figure 3.16 shows the shielding case for 0 cm of shielding and Figure 3.17 shows the case for 2 cm of shielding. In both of these cases the pressure tube assembly, as described in Section 3.3.2, can be seen with the moderator around it, and inside of the pressure tube is the shielding. The air (red) that makes up the tally volume is within the black outlines, and surrounded by the shield fill, also air. In Figure 3.17 the shield material, steel in this case (yellow), can then be seen around the air fill. It is worth noting that there is a small amount of heavy water coolant (green) around the shield, or the air in the 0 cm case. This water is left because one of the requirement of the robotic system is a minimum 10 cm$^2$ bypass flow area.
Figure 3.15: Non-Shielded Tally Arrangement

Figure 3.16: Shielded Tally Arrangement with 0 cm Shielding

Figure 3.17: Shielded Tally Arrangement with 2 cm Shielding
Chapter 4

Graphical User Interface (GUI)

MCNP is a specialized program, with a primitive input structure based upon operating system command line architecture. Given this, it often requires a specially trained person to modify, or even to run, simulations through MCNP. The end goal of this work is to generate a simulation that can predict the dose rates in the reactor, for use with an improved robotic inspection system. With this in mind, it would be beneficial if the MCNP simulation could be adjusted to a new set of inspection parameters and run, without the need of a specially trained MCNP user. With this in mind, a GUI has been developed with a non-MCNP user in mind. This GUI will allow the user to develop and run a MCNP input deck for almost any potential inspection environments, without requiring them to interact directly with the MCNP inputs. While this GUI has been developed with ease of use in mind and so that a specially trained MCNP expert is not needed, the user will still require training on the use of this GUI to ensure that the inputs and results are reasonable.

The GUI, which is shown in Figure 4.1, uses buttons and drop down menus to allow the user to select values for a number of variables. These selections are then interpreted by the GUI, and appropriate MCNP input segments are selected. These segments are
Figure 4.1: Graphical User Interface
then all arranged into an overarching folder architecture which MCNP can handle, and several auxiliary files are written. This chapter will explain the various functions of the GUI and the files, along with the file architecture that the GUI creates.

4.1 Channel Selection

The first, and arguably most important, feature of the GUI is the ability to select which channel, or channels, to empty and generate results in. This selection of channels can be done in one of two ways: either through a set of buttons arranged in the channel layout, as shown on the right side of Figure 4.1; or by entering the alpha numeric location of the channel into the channel select box, as shown in Figure 4.2. When a channel is selected by clicking the appropriate button in the reactor layout the corresponding alpha numeric location\(^1\) will also be added to the channel select box. Similarly, if an alpha numeric location is entered into channel select box, and the select channels button is pressed, it will also be marked as empty in the reactor layout. In this way, both systems can be used in conjunction, or independently, depending on which is easier for the individual user.

This channel selection mechanism allows for the user to plan the specific channels which will be emptied and inspected, thereby allowing for predictions of the dose rates and total dose that the inspection system will encounter. This is very important due to the connection between the individual channel doses, and the location, and number of surrounding channels that are also empty, given that the primary source of dose is from the irradiated fuel in the surrounding channels.

\(^1\)The alpha numeric locations are labeled with the row being referred by letter, A through Y omitting I, and the columns being referred to by number, 1 through 24
Channels that are selected to be emptied in this method have several changes made to them within the MCNP code. The first change is that the geometries for the fuel and shield plug are both removed. Additionally, the D$_2$O coolant that filled the space is modified. With the channel now clear, a number of tally volumes are entered. These tallies will be discussed in the following section.

### 4.2 Tally Location

As mentioned in the previous section, during the process of selecting the channels to be emptied and inspected, space is cleared for the tally volumes. These tally volumes take up almost the entire space within the fuel channel, and are generated for the full length regardless of the tally locations selected. There are a total of 23 separate tally volumes, each 50 cm long, which make up the full length. While the tally volumes are generated for the full length, they simply act as whatever material they are made of (determined by the shielding case). The tallying capability, which allows them to generate results, is not implemented on any individual location until the tally volumes are selected as active.
4.3 Fuel Composition

The next function to be discussed is the fuel selection menu. This drop down menu allows the user to select fuel composition, by selecting the number of days since the reactor shut down. This cooling period can make a very large difference to the activity...
Table 4.1: Total Fuel Activity for Different Fuel Terms

<table>
<thead>
<tr>
<th>Days Since Shutdown</th>
<th>Total Bundle Emission Rate ($\gamma/s$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 day</td>
<td>7.58E+16</td>
</tr>
<tr>
<td>3 days</td>
<td>4.72E+16</td>
</tr>
<tr>
<td>7 days</td>
<td>2.46E+16</td>
</tr>
<tr>
<td>10 days</td>
<td>1.79E+16</td>
</tr>
</tbody>
</table>

of the fuel, and as such can make a large difference to the dose rates that the inspection system is going to experience. There are currently four different fuel terms that the GUI can use, as shown in Table 4.1, and additional fuel terms could be added for improved accuracy. Table 4.1 shows the total activity for a single bundle of fuel.

### 4.4 Shielding Case

In addition to being able to select the channel to inspect and the location for the tallies, the user is also able to select the shielding case to simulate. As discussed in Section 3.3.7.1, the shielding cases are used to simulate the shielding on the inspection system. There are two main variables that can be changed for the shielding case: the thickness/layout of the shielding and the material. The shielding case is selected through the use of a drop down menu and an accompanying image. The menu allows for the selection and the image shows how the selected shielding arrangement will be implemented. If a shielding case with a shield is selected, the material selection drop down menu becomes available. This material menu allows for the user to select the material that will be used in the shield. Currently only three materials options are available: SS 304L, Tungsten, and Depleted Uranium, however, further options could be added as needed. Once a material is selected, the density of the selected material will be displayed below the selection menu. Figure 4.4 shows the full shielding case selection interface, including the material selection menu.
There are currently seven options for shielding case, as discussed in Section 3.3.7, which include a non-shielded case and shielded cases ranging from 0 cm to 3 cm in shield thickness, at 0.5 cm intervals. All of the shielded cases (not the non shielded case) use the same tally volume to allow for accurate comparisons between cases.

4.5 Other Features

In addition to the four main functions of the GUI, there are also several secondary functionalities that have been implemented. These functionalities include the ability to select the number of particles to be simulated, automatic and manual selection of the number of computer threads to use, and the ability to run the MCNP simulation from the GUI.

4.5.1 Number of Particles

At the bottom left corner of the GUI interface, as seen in Figure 4.1, is the selection interface for the number of particles. This allows users to enter the number of particles, in scientific notation, that they would like MCNP to run. This value changes how long the run will take, but also changes the statistical error of the simulation. The default value of $1 \times 10^9$ particles has been found to be an adequate number of par-
articles for simulating channels without shielding to achieve acceptable statistics. On shielding cases, it is recommended that users run the simulation at $1 \times 10^9$ to generate rough estimates and the associated error, and then increase the number of particles simulated, in order to decrease the error.

### 4.5.2 Number of Threads

Next to the number of particles input is the number of threads menu. This menu allows the user to run the simulation using more than one thread (virtual core). By increasing the number of threads with which to run MCNP, the speed of the simulation can be increased. The number of threads used simply tells MCNP how many of the computers logical cores to task, and, as such, can increase the speed of the simulation without affecting the results.

The drop down menu allows users to select a number of threads between one and the maximum number of the computer. If the user selects the maximum number of threads available a warning will pop up alerting the user that this may lead to unexpected issues with the computer and asking if they want to continue. This warning appears because MCNP fully tasks as many threads as it has access to, and, as such, the computer can respond slowly to other commands. The maximum number of threads available is detected by the GUI upon running it, by interfacing with constant, pre-defined Microsoft Windows environment variables. Additionally, the value for number of threads to use is set to one less than the maximum number by default, in order to maximize speed of simulation without overworking the computer.

### 4.5.3 Run MCNP

Once the user has selected all of the channels they wish to inspect, the tally locations to generate results at, the shielding case and material, the fuel composition, the
number of particles, and the number of threads with which to run the simulation on, they need to be able to generate an MCNP input deck. This is done by clicking the “Generate Input” button in the lower centre of the GUI. If the user has failed to correctly select something, such as not selecting a channel or tally, an error message will appear notifying them of this. If no errors are present, the necessary input deck is generated at a predefined location, as discussed in Section 4.6.1 below, and the “Run” button appears next to it. This run button will run the simulation using the input that was just generated.

4.6 Auxiliary Files

The GUI serves the purpose of allowing the user to interact with the MCNP files through a graphical interface. In order to allow the GUI to change parts of the MCNP input, without changing the entire file, a segmented input file was used. This allows for one main file to contain READ commands which point to the other necessary files. Additionally, the input files that are being read do not need to be in the same folder and the main file, and can be split into a neat file architecture which is easier to navigate if needed. In addition to the inputs being organized into a neat file architecture, several additional files are also generated including batch files, parsing scripts and output files. The file architecture and additional files are discussed in greater detail in the following section.

4.6.1 File Architecture

As mentioned above, a specific file architecture has been implemented to ensure that all MCNP files are in predictable locations. The GUI generates a complete file architecture, however, this would not be of any use if the location on the computer could not also be set. Given this, the GUI uses the %homedrive% and %homepath% Windows
environment variables in order to locate a suitable location to generate results in. When combined as \%homedrive\%\homepath\%, these variables direct to the folder directory for the current user, as set up in Windows (e.g., C:\Users\Jordan Gilbert\). This will work regardless of the username and default drive. From here My Documents can be accessed, and it is here that the GUI generates all files.

The first step is to generate a new folder within the My Documents folder, called DNGS Results, where all further files can be generated. Within the DNGS Results folder everything is divided between Input and Output. Figure 4.5 shows the full folder architecture and the files within each folder. The input folder contains a folder which holds the part files, the main input files (DNGSCoreDose#.txt) and the MCNPRun.bat file. The part files are broken up into sub folders, which each hold a number of different text files, each describing a piece of the MCNP input. There are five main input files, each named sequentially, 1-5. These main files are identical with the exception that they each have a different random number seed. Finally, the batch file is made to set up the temporary environment variables that MCNP needs to run, and then calls MCNP five times, once for each of the five main files. This also tells MCNP to name all of its output files output#, where the # corresponds to the input file.

The output folder contains several types of files: the parser, the output files, and the parsed output files. There are two MCNP output files: output files (output#.o) which contain the results of the MCNP simulation and runtape files (output#.r) which contain records of the simulation as it runs, in case a run is interrupted. As discussed above, the # refers to the corresponding input number. The runtape files are not overly useful once the simulation is complete and as such are not used. The output files, however, have a great deal of information contained within. They can be very confusing to read and sometimes it is beneficial to be able to separate the results
of interest from the rest of the information. The parser serves this purpose by reading through the MCNP output files and taking the dose results and Tally Fluctuation Charts (TFCs) and storing them in Comma Separated Value (CSV) files for easy use with Microsoft Excel, or other similar programs. There is one dose CSV file and one TFC CSV file for each input file (1-5).
Figure 4.5: File Architecture of Created Directories
Chapter 5

Results

Building the model and ensuring its accuracy was an iterative process. Given that the purpose of this model is to accurately predict dose to inspection systems within the reactor, the first, and possibly most important, result that needs to be examined is benchmarking. Once the model was benchmarked, and the accuracy of the dose predictions had been shown, results were generated for different configurations of shielding on the robotic inspection system. Based on the results, estimations of component lifetimes could be determined. This chapter will go through the different tests performed and explain both the scenario tested and results found.

5.1 Benchmarking

As mentioned, benchmarking the model against existing values was key in ensuring the accuracy and reliability of this tool. Proper benchmark values were extremely difficult to obtain, as the majority of measurements from within the core are either proprietary knowledge, or are neutron measurements from reactor operation. While data about the dose rates within an offline reactor is sparse, some benchmarking data points were found.
The first benchmarking data point that was found is from OPG documentation, specifically the design requirements for the current CIGAR system [15]. The requirement states the system must be able to survive in radiation fields of $1 \times 10^6$ R/hr ($8.77 \text{kGy[Air]} / \text{hr}$) [15]. This value is an operational limit, meaning that if an inspection system can withstand this dose rate it will be able to survive in the reactor. For this reason it is expected that the dose rates predicted by the model should be below this operational limit. While this is not a true benchmark value it does still provide an upper bound that the simulation results can be compared against.

In an attempt to further verify the accuracy of the dose estimates generated by the simulation, further benchmarking values were sought out. As mentioned above, this proved to be difficult and only one other value was found. This second benchmarking value comes from work done by Atomic Energy of Canada Limited (AECL) [44]. As part of a project to locate garter springs using spectroscopy, computer modeling of the offline reactor was performed. This model was not a complete reactor model as was developed for this thesis, but was instead a single lattice cell with reflective boundary conditions. The dose rate that was reported in the centre of an empty channel, 7 days after shutdown, was $560 \text{kR/hr}$ ($4.91 \text{kGy[Air]} / \text{hr}$) [44]. In addition to a benchmark value for the dose rate from fuel, a benchmark value for the activation term was also included in the paper. The dose rate from only the activation products was $180 \text{kR/hr}$ ($1.58 \text{kGy[Air]} / \text{hr}$) [44].

In order to generate a value which could be benchmarked against the two reference values found, a comparable simulation setup was prepared. The simulation was run with channel M13, one of the centre channels, emptied, and using a 7 day decayed fuel term. Tallies were set at all locations within the core ($-275 \text{cm}$ to $275 \text{cm}$), and the results were averaged to find the in-core dose rates. The simulation was run and
Table 5.1: Simulated and Benchmark Dose Rates

<table>
<thead>
<tr>
<th></th>
<th>Simulated (kGy[Air]/hr)</th>
<th>Reference (kGy[Air]/hr)</th>
<th>% Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Only</td>
<td>4.49</td>
<td>4.91 [44]</td>
<td>9.16</td>
</tr>
<tr>
<td>Activation Only(^1)</td>
<td>1.12</td>
<td>1.58 [44]</td>
<td>29.1</td>
</tr>
<tr>
<td>Total (AECL)</td>
<td>5.61</td>
<td>6.49 [44]</td>
<td>14.0</td>
</tr>
<tr>
<td>Total (OPG)</td>
<td>5.61</td>
<td>8.77 [15]</td>
<td>30.0</td>
</tr>
</tbody>
</table>

\(^1\)Note that this activation dose rate was calculated from the ratio of the activation dose rate to the total dose rate from the AECL paper, as explained in Section 3.3.6.1.

An estimated dose rate of 4.46 kGy[Air]/hr was found for the fuel alone. With the additional 25% increase to account for the activation term, the total dose rate estimate was 5.58 kGy[Air]/hr. These dose rates are presented in Table 5.1 alongside the benchmarking values.

Given that the source term used for this simulation is based upon discharge fuel, it was expected that the resulting dose estimations should be slightly higher than the benchmark values, where the burnup would be mixed. With this in mind it is key to note that the percent difference shown in Table 5.1 is likely an underestimate of the actual error. Additionally, the activation dose for the simulation is estimated using a multiplication factor, that was calculated based on the activation dose rates from the AECL paper [44], as described in Section 3.3.6.1. This means that comparing the activation dose rates is only a rough approximation. Even with these errors in mind, it can still be seen that the estimated dose values are reasonably close to the benchmarking values, especially when the uncertainty of the benchmarking values is considered. This benchmarking shows that the simulation estimates the dose rates present sufficiently well, such that the estimates can be used to predict component lifetime for the improved inspection system.
Table 5.2: Sensitivity of Dose to Bundle Power

| Bundle Power (kW) | Dose Rate (kGy|air|/hr) | % of Original Dose |
|-------------------|----------------|-------------------|
| 810               | 4.07           | 90.68%            |
| 900               | 4.49           | 100.00%           |
| 990               | 4.92           | 109.67%           |

5.2 Sensitivity Analysis

In the development of the model and the simulation a number of small simplifications were made. These simplifications, combined with the uncertainty in some of the input parameters, could result in an error in the calculated results. This error is almost impossible to quantify due to the number of uncertainties and errors that are combined within the model. To ensure that the results being presented are reliable, a sensitivity analysis has been performed. In this sensitivity analysis the fuel source term was modified in order to determine the sensitivity of the dose to the bundle power used for the source term. The bundle power was varied by ±10% and a new source term was developed. This source term was then used as an input to the model and a new set of simulations was run. The results of these simulations can be found in Table 5.2. It can be seen that a 10% change in the fuel bundle power will also result in approximately 10% change in the dose rate. This shows that the model responds in a predictable and reliable manner.

5.3 Days After Shutdown

With the benchmarking confirming that the dose rates predicted at 7 days after shutdown, it was reasonable to extend this to other decay times. Four times were selected for testing at this time, with more able to be added reasonably simply as future work. Table 5.3 shows the estimated in-core dose rates at the four testing times. Additionally, Figure 5.1 shows the trend of the dose rate, as well as the total fuel activity.
present within the reactor. It is clear that both the fuel activity and the dose rate decrease in an exponential trend, however the activity falls off much quicker, relative to the dose rate.

By being able to estimate the dose rates present within the reactor at different times after shutdown, an ideal inspection schedule can be predicted. This ideal inspection schedule would depend on the locations within the reactor, the hardness of the final electronics selected, and the amount of shielding selected. By varying these, and other variables, the delay between reactor shutdown and inspection start can be minimized, without fear of the electronics failing unexpectedly. Minimizing the delay between shutdown and inspection will allow the inspections to be completed faster, which would in turn reduce the required downtime. Reductions to downtime are always highly valued due to the large cost associated with reactor shutdown.

5.4 Axial Distribution

The dose rate within the reactor is not uniform throughout, certain areas where there are large amounts of shielding have lower dose rates from the fuel. Figure 5.2 shows the dose rate along the fuel channel, including in the end fitting. The dose rates shown do not include those from activation. This is especially important at either end of the reactor, in the end fittings, where the shield wall blocks the majority of the dose from the fuel. In these areas, the activation term will be more important, however, the current method of estimating activation dose is not deemed appropriate to use here, and as such the results for the end fittings are unreliable. The vertical lines labeled as shield wall show the approximate start of the shield walls. The tally volumes at ±300 cm from reactor centre are half in the end fitting and half in the core, thus the vastly reduced dose relative to the core dose. It is worth noting that even with
Table 5.3: Estimated Dose Rates at Different Days After Shutdown

<table>
<thead>
<tr>
<th>Days Since Shutdown</th>
<th>Estimated Dose Rate (kGy/Air/hr)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Fuel Only</td>
</tr>
<tr>
<td>1</td>
<td>6.70</td>
</tr>
<tr>
<td>3</td>
<td>5.85</td>
</tr>
<tr>
<td>7</td>
<td>4.49</td>
</tr>
<tr>
<td>10</td>
<td>3.79</td>
</tr>
</tbody>
</table>

Figure 5.1: Estimated Dose Rate and Total Fuel Activity vs. Days Since Shutdown
Figure 5.2: Axial Distribution of Dose Rate from Fuel Along Channel M13

some dose caused by activation, the end fittings have extremely low dose rates. This makes these locations ideal for placement of components that are more susceptible to failure as a result of exposure to radiation. Future work on the inspection system should look at the possibility of moving the sensitive computer components off of the inspection head, and back to the end fitting, where it will survive much longer. It is also important to note that the dose rates within the core bounds are very stable, this makes estimating total dose easier, as the entire core residence time will be at the same dose rate.
5.5 Shielding Cases

As discussed in Sections 3.3.7 and 4.4, a number of shielding configurations have been implemented into the simulation to represent the shielding that will be present on the inspection system. These shielding cases allow for the simulation to predict the dose to the key components and to allow different shielding configurations to be tested in order to find the optimal solution. Tests were done for three different shielding materials, at six different thicknesses ranging from 0.0 cm to 3.0 cm with 0.5 cm steps. The materials that were tested were stainless steel (SS304L), tungsten, and depleted uranium. Figure 5.3 shows the estimated dose rates inside of the three different material shields, and the trends that develop with thicker shields. It is worth noting that the dose rate for 0 cm of shielding is the same run, as there is no material present. The 0 cm run was included because the shielding case requires a smaller tally volume than a non-shielded case, and this smaller tally volume results in different statistical variation. In order to keep all shielded runs comparable the tally volume that was used is that of the 3 cm run, which has the smallest diameter. With all of the shielding runs, including a non-shielded case, run with this small tally volume, all of the results will be comparable and any change will be due to the addition of shielding material.

One issue that has been noted at higher shielding thicknesses is an increase in the statistical error predicted by MCNP. This is because MCNP uses large numbers of result samples to form a final estimate and the fewer point that can be sample the higher the statistical error. As the shielding increases the number of particles that make it through the shield decreases, and therefore the statistical error increases. This can be overcome by increasing the number of particles used or by implementing a bias to focus the calculations on the areas of interest.
Figure 5.3: Dose Rates for Various Shielding Materials
5.6 Estimated Component Lifetime

Each of these tested variables, time after shutdown, location within the reactor, and shielding, can be combined in order to generate a model that exactly describes the state of the reactor when inspections are planned. This estimation of the dose rates for the exact inspection environment is vital in estimating the lifetime of the inspection system. Components for the inspection system can be broken into two categories: unshielded and shielded components. Some components, such as the ultrasonic sensors, drive mechanisms, and resolvers cannot be shielded as they need to be in direct contact (line of sight for the ultrasonics) with the pressure tube. For these components, the lifetime is estimated using unshielded runs. Shielded components, such as the microprocessor and computer components, are much more radiosensitive, and as such will be shielded to increase their lifetime. These components will have lifetimes generated for multiple shielding cases and the best case will be selected. Table 5.4, duplicated here from Section 2.3.3, shows a list of the radiation hardnesses of some key components.

Table 5.4: Radiation Hardness of Various Robotic Components

<table>
<thead>
<tr>
<th>Type</th>
<th>Component</th>
<th>Total Dose</th>
</tr>
</thead>
<tbody>
<tr>
<td>Computing</td>
<td>Integrated Circuit (CMOS) [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>Microprocessor [33]</td>
<td>3 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>RAM [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>A/D Converter [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>Bus Interface [33]</td>
<td>3 kGy[Si]</td>
</tr>
<tr>
<td></td>
<td>SERializer/DESerializer [33]</td>
<td>10 kGy[Si]</td>
</tr>
<tr>
<td>Location Sensors</td>
<td>Optical Encoder (Standard) [34]</td>
<td>1 kGy</td>
</tr>
<tr>
<td></td>
<td>Optical Encoder (Hardened) [34]</td>
<td>1 MGy</td>
</tr>
<tr>
<td></td>
<td>Optical Encoder (Fibre) [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Resolver [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Potentiometer [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td>Sensors</td>
<td>Ultrasonic Sensor [34]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Distance Sensor [35]</td>
<td>20 MGy</td>
</tr>
<tr>
<td>Other Components</td>
<td>Conductors/Connectors [35]</td>
<td>10 MGy</td>
</tr>
<tr>
<td></td>
<td>Drive Mechanisms [35]</td>
<td>10 MGy</td>
</tr>
</tbody>
</table>
5.6.1 Dose Comparison

Before the simulated doses can be compared to the component hardmesses it must be ensured that the doses are in the same units. The simulation and benchmarking values are all in units of Gy[Air], meaning Gy in air, and as such can clearly be compared. The robotic components on the other hand have their hardness values measured in Gy[Si]. Some of the components do not state what material the dose is measured in so it is assumed to be silicon, as this is standard for electronics. To convert the hardmesses from Gy[Si] to Gy[Air] the following equation can be used [24]:

\[ D_{\text{Air}} = D_{\text{Si}} \frac{\mu_{\text{en}}(\text{Air})/\rho}{\mu_{\text{en}}(\text{Si})/\rho} \]  

(5.1)

Where \( D \) is the dose in the material and \( \mu_{\text{en}}(\text{material})/\rho \) is the mass energy absorption coefficient for the respective material. It can be seen that the ratio of the mass energy absorption coefficients acts as a factor by which the dose can be multiplied in order to convert it. The mass energy absorption coefficients are energy dependant so an averaged ratio was developed as shown in Table 5.5. It can be seen that the averaged mass energy absorption coefficient is only 1.08 and as such it is reasonable to directly compare doses in air to doses in silicon without the need for conversion.

5.6.2 Unshielded Components

As can be clearly seen in Table 5.4, the unshielded components are generally much more radiation resistant than the computing components. The ultrasonic sensors, resolvers, potentiometer, electrical conductors and drive mechanisms are all able to survive total doses of 10 MGy. Table 5.6 shows the dose rates, as discussed above, and the time to failure for these components. It is clear to see that the unshielded components can withstand the radiation environment for sufficient time to complete multiple inspections.
Table 5.5: Average Mass Energy Absorption Coefficient Calculations

<table>
<thead>
<tr>
<th>Energy</th>
<th>$\mu_{en}(\text{Air})/\rho$ [24]</th>
<th>$\mu_{en}(\text{Si})/\rho$ [24]</th>
<th>Ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.1</td>
<td>0.0233</td>
<td>0.0435</td>
<td>1.87</td>
</tr>
<tr>
<td>0.15</td>
<td>0.0251</td>
<td>0.0300</td>
<td>1.20</td>
</tr>
<tr>
<td>0.2</td>
<td>0.0268</td>
<td>0.0286</td>
<td>1.07</td>
</tr>
<tr>
<td>0.3</td>
<td>0.0288</td>
<td>0.0291</td>
<td>1.01</td>
</tr>
<tr>
<td>0.4</td>
<td>0.0296</td>
<td>0.0293</td>
<td>0.99</td>
</tr>
<tr>
<td>0.5</td>
<td>0.0296</td>
<td>0.0290</td>
<td>0.98</td>
</tr>
<tr>
<td>0.6</td>
<td>0.0296</td>
<td>0.0290</td>
<td>0.98</td>
</tr>
<tr>
<td>0.8</td>
<td>0.0289</td>
<td>0.0282</td>
<td>0.98</td>
</tr>
<tr>
<td>1</td>
<td>0.0280</td>
<td>0.0274</td>
<td>0.98</td>
</tr>
<tr>
<td>1.25</td>
<td>0.0268</td>
<td>0.0263</td>
<td>0.98</td>
</tr>
<tr>
<td>1.5</td>
<td>0.0256</td>
<td>0.0252</td>
<td>0.98</td>
</tr>
<tr>
<td>2</td>
<td>0.0238</td>
<td>0.0236</td>
<td>0.99</td>
</tr>
<tr>
<td>3</td>
<td>0.0211</td>
<td>0.0217</td>
<td>1.03</td>
</tr>
<tr>
<td></td>
<td>Average Ratio</td>
<td></td>
<td>1.08</td>
</tr>
</tbody>
</table>

Table 5.6: Time to Failure for Unshielded Components in Channel M13

<table>
<thead>
<tr>
<th>Days Since Shutdown</th>
<th>Estimated Dose Rate (kGy[Air]/hr)</th>
<th>Time To Failure¹ (hr)</th>
<th>Number of Channel Visits²</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>8.38</td>
<td>119</td>
<td>19.8</td>
</tr>
<tr>
<td>3</td>
<td>7.31</td>
<td>137</td>
<td>22.8</td>
</tr>
<tr>
<td>7</td>
<td>5.61</td>
<td>178</td>
<td>29.7</td>
</tr>
<tr>
<td>10</td>
<td>4.74</td>
<td>217</td>
<td>36.2</td>
</tr>
</tbody>
</table>

¹Failure time assumed to be when total dose of 10 MGy reached.
²Assuming 6 hours in core per channel visit.
5.6.3 Shielded Components

In addition to the components discussed above, that do not require shielding, there are a number of more radiosensitive components. These components will mostly be integrated circuits and microprocessors that will be used to control the motors and interpret information from the various sensors. Specific parts have not yet been sourced for the improved inspection system, and as a result the exact radiation hardness is not known. Radiation hardened components have been found and are being used as a baseline for calculating lifetimes. These lifetimes can be updated once specific components are selected.

Based on the radiation hardness of the baseline electronics, as shown in Table 5.4, it can clearly be seen that the microprocessor and bus interface will be the most radiosensitive component. Using the total dose acceptable for these components (3 kGy[Si]) as the failure dose of the system, it is possible to estimate the lifetime. A series of runs were done at 7 days after shutdown, testing each shielding material and thickness as outlined in Section 3.3.7.1 and the results are shown in Table 5.7.

Given that inspections currently take a total of approximately 12 hours to complete [48], including placement of the inspection system into the pressure tube and preparation to inspect, this means that the inspection head should only be in the reactor core for approximately 6 hours. With this in mind, any shield that does not provide a lifetime of at least 6 hours will not be acceptable. This immediately eliminates any shield made of steel or lead, and any thickness below 2.0 cm for both tungsten and depleted uranium. Tungsten could be used at 2.5 to 3.0 cm of thickness and depleted uranium could be used at thicknesses above 2.0 cm. With that being said however, most of these materials have additional factors that should be taken into consideration before selecting a shield. Depleted uranium could be activated,
potentially forming plutonium, which would have a number of drawbacks. Tungsten can lead to the generation of x-rays, potentially increasing the dose to the electronics. Lead has a mechanical issue, in that it is soft and therefore would require a structural shell to prevent pieces of the shield being scrapped off and left within the reactor. Further consideration will be required before a shield selection can be finalized.

Table 5.7: Estimated Failure Time for Shielded Components

<table>
<thead>
<tr>
<th>Thickness (cm)</th>
<th>Steel (hr)</th>
<th>Lead (hr)</th>
<th>Tungsten (hr)</th>
<th>DU (hr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>0.51</td>
<td>0.51</td>
<td>0.51</td>
<td>0.51</td>
</tr>
<tr>
<td>0.5</td>
<td>0.54</td>
<td>0.78</td>
<td>0.95</td>
<td>1.12</td>
</tr>
<tr>
<td>1.0</td>
<td>0.65</td>
<td>1.14</td>
<td>1.63</td>
<td>2.05</td>
</tr>
<tr>
<td>1.5</td>
<td>0.78</td>
<td>1.59</td>
<td>2.67</td>
<td>3.58</td>
</tr>
<tr>
<td>2.0</td>
<td>0.92</td>
<td>2.18</td>
<td>4.36</td>
<td>6.15</td>
</tr>
<tr>
<td>2.5</td>
<td>1.12</td>
<td>3.00</td>
<td>7.27</td>
<td>10.43</td>
</tr>
<tr>
<td>3.0</td>
<td>1.35</td>
<td>4.21</td>
<td>12.00</td>
<td>17.14</td>
</tr>
</tbody>
</table>

\(^1\)Failure time assumed to be when total dose of 3 kGy[Si] reached.
Chapter 6

Conclusions

This thesis has examined the development of an MCNP model for predicting the dose rates to an improved robotic inspection system, and the lifetime of the components within such dose rates. A background of the CANDU reactor and inspection systems was presented and the potential radiation effects and damage mechanisms that could affect the inspection systems were explained. In addition to the radiation damage mechanisms, a list of key components and their radiation tolerances was presented. Following this, the Monte Carlo method was explained and the MCNP model was described in detail. Additionally, the GUI, which was developed to interface with the MCNP model, was presented, including an in-depth explanation of the various functions that it can perform.

The model was then benchmarked against values from both OPG and AECL, showing good agreement with both (within 14% difference compared to the AECL benchmark). Several other tests were also described, showing the estimated dose rates for several different scenarios including: different days since shutdown, different axial locations within the reactor, and different shielding cases. Following this, the component lifetimes were estimated using the radiation tolerance and the estimated dose rates. It
was found that most components would be able to withstand well over 100 hr in channel without the need for shielding, but certain components such as the microprocessor, would only last for a limited time, even with large amounts of shielding. If these components needed to be on the inspection head they could be shielded with tungsten of thicknesses in excess of 2.5 cm, which would allow them to survive for one or two inspections. Alternatively, if the components did not need to be on the inspection head, they could be moved into the end fitting where the reactor shielding will reduce the dose rate enough to allow the components to be used for multiple inspections.

The MCNP simulation and accompanying GUI form a tool that allows for dose fields to be predicted throughout the reactor. Until now there has not been any such tool, or even static dose field data. The availability of accurate input data, such as source terms, and appropriate benchmarking values has hampered the development of the simulation, however, the completed simulation has been shown to predict the radiation fields within acceptable deviation from the benchmarking values that were found. While this simulation was developed around the Darlington reactor, the modular approach taken in its development would allow it to be easily modified to simulate any CANDU reactor.

6.1 Future Work

This simulation and the GUI that accompanies it have been shown to be a sufficiently accurate tool for the purpose of predicting dose rates within the reactor. That being said, there are still a number of ways that they could be improved. The first major improvement that could be added is an improved activation term. As discussed in Section 3.3.6.1, the activation term that is being used is an approximation and as such leaves room for further development. A separate simulation could be performed
to develop an explicit activation source term, which would include the location and isotopic activity of any activation products. This activation source term could then be implemented as an input to the overall radiation field model and would improve the accuracy of the dose predictions.

In addition to a proper activation term, further benchmarking is required to ensure the complete accuracy of the predicted doses. Ideally this benchmark would be against a measured value from a reactor. This benchmarking is especially important for the end fittings, where a large portion of the inspection systems will be housed, but cannot be accurately modeled yet. Further improvements to the fuel source term are also possible. The current model uses discharge fuel as a source term, which should result in higher dose rates as compared to a mixed burnup fuel term.

Beyond these more important improvements, there are a number of less important changes that could be made to further improve accuracy. The current model focuses purely on photons, however, photons will lead to the development of beta particles. These more complex phenomenon would likely cause the simulation to be significantly slower and as such may be best implemented as a separate simulation on a single channel basis. In addition to the changes to the MCNP simulation, there are also some improvements that can be made to the GUI. These improvements would include determining recommended run times for different simulations, thereby ensuring the statistical error is sufficiently low. Additionally, improved output processing would allow the user to view data from the simulation in a more manageable manner. Finally, more information is needed for the radiation hardness of the specific components that will be used for the improved inspection system, once these components have been selected.
References


Appendix A

OPG Drawings

Table A.1: OPG Technical Drawings Used [41]

<table>
<thead>
<tr>
<th>Drawing Title</th>
<th>Drawing Number</th>
</tr>
</thead>
<tbody>
<tr>
<td>End Fitting Assy.</td>
<td>NK38-GEN-31120-0001-001</td>
</tr>
<tr>
<td>Fuel Channel End Fitting Finished Machined Body Detail</td>
<td>NK38-DFN-31120-0002-001</td>
</tr>
<tr>
<td>Fuel Channel Assemblies Major Assembly Installation Requirements</td>
<td>NK38-GEN-31100-0003-A01</td>
</tr>
<tr>
<td>Fuel Channel Assembly Internal End Fitting Components Control Dimensions</td>
<td>NK38-GEK-31100-0009</td>
</tr>
<tr>
<td>Fuel Channel Assembly Internal End Fitting Components Assembly</td>
<td>NK38-GEK-31100-0006</td>
</tr>
<tr>
<td>Fuel Channel Assembly Inlet Shield Plug Body Detail</td>
<td>NK38-DEK-31130-0002</td>
</tr>
</tbody>
</table>
Appendix B

Activated Reactor Components

The source term that describes the activation products within an offline CANDU reactor is very complicated. Every component within the reactor core will become activated over the reactor operation, and therefore produce radiation. The degree of activation is a function of the total operation time of the reactor, the location of the component being activated, and of the the material that is being activated. Given this, it is very difficult to generate an accurate activation source term. One method to get around these issues would be to generate the activation source term for end of reactor lifetime, right after shutdown. This would result in the highest activity for the activated components, therefor resulting in a conservative dose estimate.

The other issue that arises is the location to generate the source term in within the model. Every component within the reactor will be activate, some more than others, and there is limits on how many source locations can be defined. This becomes even more complicated in cases where a repeated geometry is used, such as in this model. In order to get around this, certain activation components need to be ignored and others need to be combined.
An activation term was generated for the current model in an attempt to accurately predict the dose rates caused by the activation products, however a lack of accurate and reliable input data resulted in the failure of this attempt. As discussed in Section 6.1, there are several ways to improve the activation source term, however these are future work.

For the activation term that was generated, the source term used was based on a decommissioning activation term as found in [50]. This paper used MCNP and ORIGEN2 in order to develop activity amounts in each separate component. Unfortunately, this work was for a CANDU 6, which has different dimensions, and only has 380 channels. This discrepancy was accounted for by using the ratio of channels to proportionally increase the activities. While this source term was not ideal, it was the best one that could be found and as such was used.

The location of the source term also proved difficult. As mentioned earlier in this section there are limits on the number of source locations that can be used. The source locations that were used for this attempt can be broken down into four categories: the pressure/calandria tubes; the SSC, such as the shield wall, end fitting, and shield plugs; the reactivity devices; and the calandria tank.

The calandria and pressure tubes were the first to be added as they are directly beside both the fuel and the location of interest. In order to reduce the number of separate source locations needed the calandria tube source term was merged into the pressure tube source term. This should have little effect on the results, as the difference in shielding will be minuscule.

The SSC were the next location that was developed. The SSCs are the components
around either side of the reactor, such as the end fitting assembly and the shield wall. The shield plugs and liner tubes within the end fittings were modified in order to make them all the same cell as the end fitting. Given that the material for all of these components is the same it will not affect the results. It does however allow the entire area to be defined as a single location. The shield wall could then be defined separately, as it uses different materials.

The reactivity devices, or control mechanisms, were the next, and most difficult, activation product location added. The control mechanisms were not included in the initial model of the core due to the small effect they should have on a gamma shielding case. They are however very important when it comes to neutron shielding, and as such they can be very highly activated. Additionally, the reactivity devices do not appear in a pattern that lends itself to being modeled through repeated structures. This was avoided by adding several, semi-circular, vertical extrusions to the moderator volume between the channels, on either side of the lattice cell. These extrusions lined up with the next lattice to form vertical cylinders. The cylinders were placed in locations that should average the true locations of the reactivity devices, but without requiring separate structures. These cylinders were made of heavy water, as the amount of actual metal present should not provide a large amount of shielding. This was only meant to be a temporary placeholder, and further work would be required to properly model this complex system. While this was a rudimentary approach it did provide appropriate locations to place the activation term.

The final component that needed an activation term was the calandria tank around the outside, however it was very straightforward to implement. Unfortunately, even with all of the component locations and the source term, the results were extremely low, and did not match benchmark values. Given that more accurate input information
was not readily accessible, and that time was short, this work was determined to not be worth pursuing for this thesis, and as such would be marked as future work.