IMPROVEMENTS TO THE PSBR FUEL MANAGEMENT AND NEUTRONICS MODEL IN SUPPORT OF THE NEW CORE MODERATOR ASSEMBLY

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Andrew Bascom

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The dissertation of Andrew Bascom was reviewed and approved* by the following:

Kenan Ünlü  
Professor of Mechanical and Nuclear Engineering  
Director of Radiation Science and Engineering Center  
Dissertation Co-Advisor  
Co-Chair of Committee

William Walters  
Assistant Professor of Nuclear Engineering  
Dissertation Co-Advisor  
Co-Chair of Committee

Azaree Lintereur  
Assistant Professor of Nuclear Engineering

Joshua Robinson  
Associate Professor of Materials Science and Engineering

Brenden Heidrich  
Chief Scientist, Irradiation  
Nuclear Science User Facilities  
Idaho National Laboratory  
Special Member

Arthur Motta  
Chair of Nuclear Engineering  
Professor of Nuclear Engineering and Materials Science and Engineering

*Signatures are on file in the Graduate School
Abstract

Improved software for the modeling of the Penn State Breazeale Reactor (PSBR) has been developed to accommodate newly installed experimental equipment with an impact upon reactivity and in-core power distribution. A new reactor tower, moderator assembly, and neutron beamports were installed at the Penn State Breazeale Reactor (PSBR) during the summer of 2018, which the existing fuel management model was unable to support. This thesis will discuss the changes made to the fuel management and experimental analysis models in use at the PSBR to accommodate the new experimental facilities. In addition to changes made to allow the model to continue to perform its original functions, improvements that allow simulations to be performed faster and with less risk of human error will be discussed. These improvements include a set of variance reduction parameters generated using a custom quadrature set, and a program to automatically shuffle the core loading to achieve a user-specified reactivity change.

The PSBR is a TRIGA Mk III research reactor licensed to operate at 1MW thermal. The reactor uses a mixture of both 8.5 and 12 wt% Uranium-Zirconium Hydride fuel with a complex burnup history dating back to 1965. An in-house code, TRIGSIMS, is used to track the burnup of this fuel and predict power distribution and reactivity effects for changes in core loading by coupling the MCNP and SCALE codes. The recently installed core moderator assembly has a complex geometry that cannot be modeled by the TRIGSIMS code. Rather than implementing changes within the TRIGSIMS source code, a script was prepared to implement the necessary geometry definitions directly in the MCNP input generated by TRIGSIMS. This allows the new core moderator assembly to be analyzed independently of fuel burnup calculations.
After adding the capability to model the new moderator assembly, tools were prepared to streamline the analysis process of new core loadings. The preparation of input decks, previously a manual and time-consuming process, was automated such that only a single input deck need be prepared for a new core loading pattern. The code then runs all cases and automatically produces a single output report taking the place of dozens of output files which required manual processing.

To accelerate the process of simulating experiments using the new beamports, a set of variance reduction parameters have been generated using the ADVANTG code. The low flux within the beamtubes relative to the core region makes analog simulations difficult and time consuming. Extensive testing was performed using both built in and custom quadrature sets in order to achieve the greatest speedup within Monte-Carlo simulations. The variance reduction technique employed utilizes an estimate of the adjoint flux to define weight windows within MCNP that will preferentially transport particles along the beamtube of interest within a simulation without biasing the result. In addition to increasing the speed of simulations by a factor of up to 96, this weight window based variance reduction can be easily applied to new problems with only minimal modifications to inputs. This allows it to be utilized by PSBR staff and graduate students without the need to commit significant amounts of time to learning variance reduction techniques.

A new method of designing core loading patterns will be presented to aid in future fuel management efforts. Software to perform semi random shuffles was written and coupled with an existing capability of TRIGSIMS to estimate reactivity changes using the ADMARCH diffusion code. This approach has been shown to be able to identify loading
changes capable of increasing excess reactivity of the core while complying with user
specified limits upon reactivity changes and fuel element placement. An example of its
use in core design and a plan to use this method in the design of future core loadings will
be presented.

Analysis of multiple core loading patterns using the new software will be
presented. The core loading installed at the time this work was started will be shown to
be nominally acceptable for use with the new core moderator assembly, but with little
margin for uncertainty. Development of a new core loading, Core 58A, for first use with
the core moderator assembly will be presented. This core loading was designed to
provide a greater margin for uncertainties in model predictions and allow for compliance
with PSBR license requirements in the event that the model underestimated core
parameters. Based on the agreement of model predictions and measurements for Core
58A, a plan for development of a new core loading will be presented.

Through this work, the fuel management software used at the PSBR has been
updated to allow for modeling of new experimental facilities and to streamline the
process of designing and analyzing new core loading patterns. Using a new method to
adapt quadrature sets to the recently installed beamports, adjoint driven weight window
variance reduction techniques have been applied to the PSBR model. Extensive testing
has demonstrated that this method allows Monte-Carlo simulations of the PSBR to be
performed up to 96 times faster without biasing the results. The software and variance
reduction parameter libraries produced are expected to remain in use at the PSBR for
years to come.
# Table of Contents

List of Figures.................................................................................................................ix

List of Tables .................................................................................................................... xvi

Chapter 1. **Introduction** ............................................................................................... 1

  - Section 1.1- Description of the PSBR ..................................................................... 2
  - Section 1.2- Description of the Previous D₂O Tank .............................................. 6
  - Section 1.3- Description of Other Experimental Facilities ............................... 9
    - Section 1.3.1: The Fast Neutron Irradiator ..................................................... 11
    - Section 1.3.2: The Fast Flux Tube .................................................................. 12
    - Section 1.3.3: In Core Dry Tubes .................................................................... 13
    - Section 1.3.4: The Central Thimble ................................................................ 15
    - Section 1.3.5: Ex-Core Irradiation Positions .................................................... 15
    - Section 1.3.6: The Pneumatic Transfer System ............................................... 15
  - Section 1.4- Description of the New Core Moderator Assembly ..................... 16
  - Section 1.5- Fuel Management Before the New Core Moderator Assembly ......... 23
  - Section 1.6- Core Position Identification Schemes ............................................. 28

Chapter 2. **Verification and Reconstruction of the Neutronics Model for the**

New Core Moderator Assembly ...................................................................................... 33

  - Section 2.1- Description of Previous Modeling Techniques ............................... 33
  - Section 2.2- Requirements and Design Process for an Updated Model .............. 36
  - Section 2.3- Features of the Final Updated PSBR Model ..................................... 42

Chapter 3. **Variance Reduction Techniques Employed** ................................................. 46

  - Section 3.1- Review of Previously Employed Variance Reduction Techniques ...... 46
Section 3.2- Weight Window Variance Reduction Techniques ............................................. 49

Section 3.2.1: Built in Weight Window Generator ................................................................................. 51

Section 3.2.2: ADVANTG Weight Window Generator .............................................................................. 54

Section 3.3: Testing of Variance Reduction Parameters to Generate a Final Weight Window Library ................................................................................................................................................. 63

Section 3.3.1: Use of ADVANTG’s Built in Quadrature Sets ................................................................. 64

Section 3.3.2: Testing of Custom Quadrature Sets ................................................................................. 68

Section 3.4: Generation of a Final Set of Weight Window Libraries for the New Neutron Beamports ......................................................................................................................................... 76

Section 3.5 Weight Window Libraries for the FNI and FFT ................................................................. 86

Chapter 4. New Software Tools for Modeling of the New D₂O Tank and Streamline Analysis of New Core Loadings ......................................................................................................................................... 88

Section 4.1: Subroutines Prepared to Automate Existing Core Analysis Requirements ......................................................................................................................................... 88

Section 4.1.1: Post_Trigsims ......................................................................................................................... 88

Section 4.1.2: Converting to Fixed Source ................................................................................................. 90

Section 4.1.3: Geometry Addition ............................................................................................................. 91

Section 4.1.4: Processing TRIGSIMS tallies outside of TRIGSIMS ......................................................... 92

Section 4.1.5: Setting up and Running TRIGSIMS and MCNP cases ..................................................... 94

Section 4.1.6: Loading Analysis Report Generation .................................................................................. 95

Section 4.1.7: Graphical Display of Core Loading Pattern ....................................................................... 96

Section 4.1.8: Rapid Plotting of TRIGSIMS generated MCNP Input Geometry ................................... 99

Section 4.2: Automatic Generation of New Core Loading Patterns Through Random Shuffling ................................................................................................................................. 100
Chapter 5. Development and Analysis of a New Core Loading Pattern for Operations at the New Core Moderator Assembly .............................................. 111

Section 5.1: Analysis of New Moderator Assembly with Core 57 .......................... 111

Section 5.1.1: Power Distribution ............................................................................. 112
Section 5.1.2: D_2O Tank Worth............................................................................... 113
Section 5.1.3: Excess Reactivity ................................................................................ 113
Section 5.1.4: Shutdown Margin ............................................................................... 114

Section 5.2: Design of Core 58 for First Criticality at the New D_2O Tank .......... 115

Section 5.2.1: Core 58 ......................................................................................... 116
Section 5.2.2: Core 58A ....................................................................................... 118

Section 5.3: Development of Core Loading 59 Using New Software Tools .......... 122

Chapter 6. Summary, Conclusions, and Recommendations ............................. 126

Section 6.1: Summary ............................................................................................ 126
Section 6.2: Conclusions ...................................................................................... 127
Section 6.3: Recommendations for Future Work .................................................. 130

Works Cited ............................................................................................................ 133

Appendix A: Example ADVANTG Input Deck ....................................................... 137

Appendix B: Additional Figures ............................................................................ 138

Appendix B.1: Adjoint Flux Maps ......................................................................... 139
Appendix B.2: Forward Flux and Response Density Maps .................................... 146
Appendix B.3: Weight Window Map Sets .............................................................. 152
Appendix B.4: Collapsed Weight Window Map Sets ............................................. 158

Appendix C: Additional Tables ............................................................................... 163
List of Figures

Figure 1-1: Core Loading 57 Map colored by fuel element weight percent ............................ 3

Figure 1-2: Relative heights of fuel and control rods to reactor structural plates (SAR
     Figure 4-6 [1]) ............................................................................................................. 4

Figure 1-3: Locations of safety system and auxiliary detectors relative to reactor core
     (SAR Figure 4-5 [1]) ....................................................................................................... 6

Figure 1-4: Layout of previous beamport paths relative to core and D2O tank (Uçar
     Figure 3-6 [5]) ............................................................................................................. 7

Figure 1-5: Relative heights of previous beamports view from Neutron Beamlab (Uçar
     Figure 3-4 [5]) ............................................................................................................. 8

Figure 1-6: MTR Reactor Tower (Left) and Original TRIGA Tower (Right) with original
     beamports. ....................................................................................................................... 9

Figure 1-7: Map of pool and experimental facilities with reactor shown in open pool “R1”
     position ......................................................................................................................... 10

Figure 1-8: Fast Neutron Irradiator (“FNI”) geometry with materials labeled. Inner
     Diameter of the FNI is 10 inches. .................................................................................. 11

Figure 1-9: Diagram of FFT Geometry showing shielding materials ................................. 13

Figure 1-10: Change in dry tube positions shown on Core 58A map ................................. 14

Figure 1-11: New core moderator assembly and rector core geometry model. No
     structural materials are included in this model (Uçar Figure 5-3 [5]) ......................... 16

Figure 1-12: Beamtube flux optimization model used by Uçar with New Beamports
     labeled (Uçar Figure 5-9) .............................................................................................. 17
Figure 1-13: Side view and 3D render of New Core Moderator Assembly showing height of new beamports relative to core structure. ................................................................. 18

Figure 1-14: Old D₂O Tank in drained south end of pool before removal ...................... 19

Figure 1-15: Core drilling from South Pool to remove old beamtubes and create openings for New Beamports ............................................................................................ 20

Figure 1-16: New Beamport openings in Neutron Beamlab .............................................. 20

Figure 1-17: New Core Moderator Assembly in south end of pool during summer 2018 pool drain .......................................................................................................................... 21

Figure 1-18: Reactor operating at New Core Moderator Assembly ................................. 23

Figure 1-19: TRIGSIMS burnup scheme flowchart [10] .................................................. 24

Figure 1-20: TRIGSIMS model of Core 57 (left) and Core 57 with the old D₂O tank (right) .................................................................................................................................. 28

Figure 1-21: Core position descriptions (adapted from SAR Figure 4-5 [1])...................... 29

Figure 1-22: Core Row/Column coordinate system map. .................................................. 30

Figure 1-23: Core ring based naming scheme .................................................................... 31

Figure 1-24: Triangular lattice coordinate system map [10] .............................................. 32

Figure 2-1: Final model of New Core-Moderator Assembly from Uçar (Figure 5-23 [5]) ................................................................................................................................. 34

Figure 2-2: PSBR Core Loading 53G [13] .......................................................................... 35

Figure 2-3: Core Loading 57 Geometry Produced by TRIGSIMS. Fuel element color in this plot is based on material ID number tracked within the model. ......................... 38
Figure 2-4: Preliminary model of the New Core Moderator Assembly including structural elements of the new D\textsubscript{2}O tank used to predict reactivity and power distribution effects upon Core 57. ................................................................. 39

Figure 2-5: Early version of full pool model showing the reactor core coupled to the new D\textsubscript{2}O Tank. ........................................................................................................................................ 41

Figure 2-6: Final full pool model including Core 58A ................................................................. 43

Figure 2-7: Tank sub-model of Core 58A ..................................................................................... 44

Figure 2-8: Open pool sub-model of Core 58A ........................................................................... 45

Figure 2-9: FFT and FNI sub-models of Core 58A ................................................................. 45

Figure 3-1: Simplified weight window splitting explanation from MCNP 5 manual [9]. 51

Figure 3-2: Results of MCNP's built in weight window generator for a tally in New Beamport 2 showing that the built in generator creates an incomplete mesh. .... 52

Figure 3-3: Results of MCNP's built in weight window generator for a point detector tally in New Beamport 2 ........................................................................................................... 53

Figure 3-4: Results of ADVANTG weight window generator for tallies in all new beamports showing weight windows covering the entire geometry. Particles will be preferentially transported towards regions with a lower weight window cutoff. .... 56

Figure 3-5: Coarse weight window mesh overlaid on core and moderator geometry ...... 57

Figure 3-6: Weight windows generated by ADVANTG code with default quadrature scheme (QR) of Order 20 showing no clearly identified streaming path along beamtubes. ........................................................................................................... 60
Figure 3-7: Weight windows generated by ADVANTG with denovo_quad_order 20 and 200^3 Nodes showing a clearly identified streaming path (in dark blue) for each beamtube

Figure 3-8: Results of varying quadrature order on speedup when generating weight windows

Figure 3-9: Adjoint flux maps for NBP2 (top) and NBP3 (bottom) generated by ADVANTG using Quadrature Order 30

Figure 3-10: Coordinate system for quadrature sets [21]

Figure 3-11: Single octant of quadrature for original quadrature set (left) and NBP3 custom quadrature set showing added vector in red (right)

Figure 3-12: Adjoint flux for NBP1 generated by ADVANTG using the unmodified Legendre-Chebyshev S6 quadrature

Figure 3-13: Adjoint flux for NBP1 generated by ADVANTG using the custom quadrature set

Figure 3-14: Results of varying MXSPLN on speedup factor

Figure 3-15: Adjoint flux for NBP2 calculated by DENOVO

Figure 3-16: Final weight window set for NBP2

Figure 3-17: Forward flux calculated by DENOVO

Figure 3-18: Response density for NBP2 calculated by DENOVO

Figure 3-19: Collapsed weight windows for NBP2 generated by ADVANTG

Figure 3-20: 1MW flux predictions for all beamports with 1 standard deviation error bars
Figure 3-21: Figure of Merit for reference, weight window, and collapsed weight window cases ................................................................. 83

Figure 3-22: Speedup factors for weight window and collapsed weight windows cases . 85

Figure 3-23: Speedup factors for higher quadrature order tests comparing built in QR S_{30}, Legendre-Chebyshev S_6 Custom, Legendre-Chebyshev S_{30}, and Legendre-Chebyshev S_{30} Custom quadrature sets ................................................................................................. 86

Figure 4-1: Post TRIGSIMS flowchart showing generation of burned input deck for existing loading pattern .................................................................................................................................................. 89

Figure 4-2: Flowchart of software workflow modifying TRIGSIMS execution [10] ...... 93

Figure 4-3: Core diagram section of OSTR Status Board Display before modifications to show PSBR geometry .................................................................................................................................................. 97

Figure 4-4: PSBR Status Board Display showing Core Loading 58A. Fuel elements are identified by PSBR element serial number .................................................................................................................................................. 98

Figure 4-5: Random shuffle algorithm flowchart .................................................................................................................................................. 103

Figure 4-6: Core Loading 58A (top) and an example core map generated by applying the random shuffle algorithm (bottom) to Core 58A. Fuel elements are identified by PSBR element serial number .................................................................................................................................................. 106

Figure 4-7: Reactivity increases for random core shuffle example showing only fuel moves that fall within user specified range .................................................................................................................................................. 107

Figure 4-8: Net reactivity changes of all iterations for random shuffle example ........ 108

Figure 4-9: Distribution of reactivity changes produced by random shuffle algorithm . 109
Figure 5-1: Map of Core Loading 57 showing ratio of power production per fuel element (NP) at New D₂O Tank vs. Open Pool showing a shift towards the front and outside of the core. ................................................................. 112

Figure 5-2: Core Loading 58 Map ................................................................. 117

Figure 5-3: Core Loading 58A Map ............................................................. 119

Figure 5-4: New reactor tower with dummy fuel elements coupled to New D₂O Tank to verify clearances during installation ................................................................. 120

Figure 5-5: Measured clearances between tank and fuel elements for all sides of the New D₂O Tank ................................................................................................. 121

Figure 5-6: Core 59 design flowchart ............................................................ 125

Figure B-1: Adjoint Flux Map for NBP1 ........................................................ 139

Figure B-2: Adjoint Flux Map for Cold Neutron Beam Port ............................ 140

Figure B-3: Adjoint Flux Map for NBP2 ........................................................ 141

Figure B-4: Adjoint Flux Map for NBP3 ........................................................ 142

Figure B-5: Adjoint Flux Map for NBP4 ........................................................ 143

Figure B-6: Adjoint Flux Map for the Reactor at the FNI (with MCNP geometry plot for context) ............................................................................................................ 144

Figure B-7: Adjoint Flux Map for the Reactor at the FFT (with MCNP geometry plot for context) ............................................................................................................. 145

Figure B-8: Forward Flux Map for New Beamports ........................................ 146

Figure B-9: Response Density Map for NBP1 ................................................ 147

Figure B-10: Response Density Map for Cold Neutron Beam Port ................. 148

Figure B-11: Response Density Map for NBP2 .............................................. 149
Figure B-12: Response Density Map for NBP3 ................................................................. 150
Figure B-13: Response Density Map for NBP4 ................................................................. 151
Figure B-14: Base MCNP Geometry for Tank Sub-model .................................................. 152
Figure B-15: Weight Window Map for NBP1 Generated by ADVANTG ...................... 153
Figure B-16: Weight Window Map for Cold Neutron Beamport Generated by ADVANTG .................................................................................................................. 154
Figure B-17: Weight Window Map for NBP2 Generated by ADVANTG ...................... 155
Figure B-18: Weight Window Map for NBP3 Generated by ADVANTG ...................... 156
Figure B-19: Weight Window Map for NBP4 Generated by ADVANTG ...................... 157
Figure B-20: Collapsed Weight Window Map for NBP1 Generated by ADVANTG ... 158
Figure B-21: Collapsed Weight Window Map for Cold Neutron Beamport Generated by ADVANTG .................................................................................................................. 159
Figure B-22: Collapsed Weight Window Map for NBP2 Generated by ADVANTG ... 160
Figure B-23: Collapsed Weight Window Map for NBP3 Generated by ADVANTG ... 161
Figure B-24: Collapsed Weight Window Map for NBP4 Generated by ADVANTG ... 162
List of Tables

Table 1-1: Summary of analysis cases for Core Loading 58 [11] .......................... 26
Table 3-1: Summary of preliminary results for full pool model simulations .............. 63
Table 3-2: Ratio of errors and FOM for preliminary full pool model simulations (Weight
Windows/Standard)........................................................................................................ 63
Table 3-3: Original and modified weights for NBP3 quadrature set ......................... 70
Table 3-4: Results of Even Moment Condition Test for NBP1 quadrature based on
Legendre-Chebyshev S6 quadrature.............................................................................. 74
Table 3-5: Summary of speedup factors for weight window simulations .................. 84
Table 3-6: Tally results and speedups for FNI and FFT ............................................. 87
Table 4-1: Summary of cases to be analyzed for new core loadings ....................... 94
Table 4-2: Summary of shuffled elements and reactivity changes ............................. 105
Table 5-1: Model keff results for all rods fully withdrawn comparing Core 57 in the Open
Pool and D2O Tank Positions ....................................................................................... 113
Table 5-2: Model keff results for safety rod at upper limit, other rods SCRAMed
comparing Core 57 in the Open Pool and D2O Tank Positions ................................. 114
Table 5-3: Modeled and measured shutdown margins for Core Loading 57 .............. 115
Table 5-4: Predicted and measured reactivity values for New D2O Tank ................. 122
Table C-1: Detailed Summary of Runtime Data for Weight Window Testing ............ 163
Table C-2: Example Loading Report after Importing to Excel ................................. 164
Table C-3: Full Cosine and Weight Data for NBP1 Quadrature Set Octant ............... 165
Table C-4: Cosines for Adding New Beamports to Quadrature Set Octant ............... 165
Chapter 1. Introduction

This work will present several upgrades and novel methods implemented to enhance the modeling capabilities to the Penn State Breazeale Reactor (PSBR). An improved neutronics model is needed to support fuel management and analysis of ongoing experiments after the installation of a new core-moderator assembly occurred in the summer of 2018. An updated model will be introduced which maintains the current methods for the analysis of new core loadings, while at the same time adding the ability to include new experimental equipment. In addition to addressing the updates necessary to support new experimental facilities, variance reduction techniques are being investigated which could significantly decrease the difficulty of modeling both new and existing facilities.

Chapter 1 will primarily discuss background information. First, the PSBR will be introduced. Next, the existing core moderator assembly and its limitations will be discussed. Other experimental facilities such as the dry tubes, FNI, and Central Thimble which currently exist and will remain in use after the installation of the new core moderator assembly, will be introduced here and will be referenced throughout this work. Having introduced both the reactor, and current experimental facilities, the last section of this chapter will introduce the new core moderator assembly.

The model of the new core moderator, both that used in the initial design and that updated to include further structural details, will be discussed in Chapter 2. Chapter 3 will focus on the use of variance reduction techniques in order to obtain model predictions within reasonable timeframes. Chapter 4 will discuss the new software developed to automate and streamline the analysis of core loadings, including a new automated core loading optimization
algorithm. Chapter 5 will demonstrate application of the updated model and newly-developed software to an existing core loading and its use in new core loadings. Finally, Chapter 6 will summarize the simulation work that has been accomplished since the new core moderator assembly was initially designed, present results of the updated model.

Section 1.1- Description of the PSBR

The Penn State Breazeale Reactor (PSBR) is a TRIGA Mark III research reactor on the campus of The Pennsylvania State University (PSU). Originally constructed in 1955 as a Materials Testing Reactor (MTR) licensed for 100kW steady state operation and using High Enriched Uranium (HEU) fuel, the reactor was converted to the current TRIGA (Training-Research-Isotopes-General Atomic) design in 1965 [1]. The core of the PSBR consists of roughly 100 TRIGA fuel elements and 4 control rods. Each fuel element contains a mixture of Low Enriched Uranium (LEU) and zirconium hydride. The uranium is enriched to just under 20% $^{235}$U. Two different weightings of uranium are used, 8.5 wt% U and 12 wt% U, with the remainder of the fuel mass being zirconium hydride metal. The active length of the fuel is 15 inches, with a graphite reflector at the top and bottom. Fuel elements are clad in 0.020 inches of 304L stainless steel [2].

Each core loading is required to contain at least one Instrumented Fuel Element (IFE). An IFE has two differences from a standard fuel element. Firstly, there are three thermocouples embedded within the fuel. One thermocouple is positioned at the axial centerline of the fuel. The others are positioned one inch above and below the axial centerline. The second difference between an IFE and a standard fuel element is the connector at the top of the element. While standard fuel has an adapter to allow the fuel to be picked up by a fuel-handling tool, the IFE
elements have a connector poll that extends to the surface and provides a conduit for the thermocouple wires. This connector extension limits where the IFE can be placed within the core, specifically in proximity to experimental equipment such as dry tubes and locations near control rods. By embedding thermocouples directly in the fuel, the peak fuel temperature in the element can be measured directly. In designing a core loading, it is generally desired for an instrumented element to be in the position of Maximum Elemental Power Density (MEPD) within the core.

The PSBR has four control rods, shown in green in Figure 1-1. Each control rod has a stroke of 15 inches, the same as the active length of the fuel. Three of these control rods, the Safety, Shim, and Regulating Rods, are Fuel Follower Control Rods (FFCRs). In addition to a 15-inch neutron absorber, these control rods contain a 15-inch section of 8.5wt/o TRIGA fuel,
which is introduced to the core as the absorber is removed. By replacing the absorber material with fuel, the reactivity worth of the control rods is increased. These control rods are connected to a drive system at the surface of the pool. The control rod drive mechanisms utilize an electromagnet to couple the control rod shaft to a motor. When the electromagnet is de-energized, the control rod falls to the fully inserted position and is suspend from the lower end of the drive mechanism. When fully inserted in the core, the FFCRs extend below the fuel and pass through an opening in the lower grid plate. Along with the top grid plate, the lower grid plate maintains alignment of the control rods. Below the lower grid plate, the safety plate serves to prevent the control rods from falling to the pool floor in the event of a mechanical failure of the control rod drive system.

Figure 1-2: Relative heights of fuel and control rods to reactor structural plates (SAR Figure 4-6 [1])
Reactor power is monitored by 5 detectors positioned outside of the core region, as shown in Figure 1-3. Two Fission Chambers (FC) are positioned on the left of the detector rack, with the second of these being connected to the Reactor Safety System’s (RSS) SCRAM circuitry as the primary indication of reactor power, and the other as a spare. A Compensated Ion Chamber (CIC) is used as a spare power channel. On the right of the detector rack sits the RSS Gamma Ion Chamber (GIC) that is used as a power range monitor for the RSS SCRAM logic, functioning as an independent high power SCRAM signal from the FC. A spare GIC is positioned above the core and serves as an auxiliary power channel. Because the detectors are positioned near the rear of the core, any change in reactivity that shifts power production towards or away from the front of the core can cause a bias in the indicated power. Performing a thermal power calibration in accordance with RSEC procedure CCP-2 with the reactor coupled to the experimental facility with the greatest positive reactivity, ensuring that the instrumentation always reads at least the actual thermal power, compensates this for. With the instrumentation calibrated at the previous D\textsubscript{2}O tank, the actual thermal power in the open pool (R1) position was 960 kW when the reactor console measures 1 MW. [3]
Figure 1-3: Locations of safety system and auxiliary detectors relative to reactor core (SAR Figure 4-5 [1])

Section 1.2- Description of the Previous D\textsubscript{2}O Tank

The previously installed D\textsubscript{2}O tank, shown in Figure 1-4, was a cylindrical tank of diameter 24 inches and thickness 12 inches. The reactor, which must be rotated 90 degrees counterclockwise, couples to the tank from the east side. The beamtube of Beamport 4 extends into the tank in the radial direct from the south. A graphite reflector measuring 9 inches by 24 inches on the west side of the tank was used to couple Beamport 7 to the system [4]. The tangential design between Beamport 4 and the core was utilized to eliminate a direct streaming
path for gamma rays from the core. Unfortunately, this introduces a significant source of gamma rays from the absorption of fission neutrons by hydrogen in the pool, with the gamma rays then streaming down Beamport 4 as illustrated in Figure 1-4.

When the D₂O tank was installed in 1997, only Beamport 4 was coupled to the tank. As can be seen in Figure 1-4, the other beamports do not align with the tank, making simple modifications to bring other beamports into service difficult. An additional reflector was necessary to couple Beamport 7 to the tank when the beamport was commissioned in 1998. This poor alignment is further complicated when looked at in the vertical direction. Figure 1-5 shows the relative heights of each beamport as well as the reactor fuel. As can be seen, only Beamport
4 lines up with the axial centerline of the fuel. Beamports 1, 3, 5, and 7 are aligned with the fuel below the axial centerline, while Beamports 2 and 6 are completely below the fuel.

![Diagram showing relative heights of previous beamports view from Neutron Beamlab (Uçar Figure 3-4 [5])](image)

The cause of the axial misalignment between the core and beamports is the transition from the MTR to TRIGA design in 1965. The structure of the MTR tower, shown in the left of Figure 1-6, allowed all beamports to point at the fuel. The fuel follower control rods of the TRIGA core extend below the level of the fuel down to the safety plate. The right of Figure 1-6 shows the tower of the TRIGA core in front of the beamports during its installation in 1965. Since the fuel follower sections of the control rods extend beneath the fueled region of the core, the active region of the TRIGA core is a greater distance above the pool floor than the original MTR design.
Section 1.3- Description of Other Experimental Facilities

In addition to the D₂O tank, the PSBR houses several other experimental facilities. The FNI, FFT, Dry Tubes, and Central Thimble are permanently installed fixtures. Locations of the experimental facilities are shown in Figure 1-7. The Pneumatic Transfer System and Ex-Core Tubes are portable and can be easily relocated to suit various experiments. Each of these facilities is briefly discussed in this section.
Figure 1-7: Map of pool and experimental facilities with reactor shown in open pool “R1” position
Section 1.3.1: The Fast Neutron Irradiator

The Fast Neutron Irradiator (FNI) is a 10-inch diameter vertical tube in the northwest corner of the reactor pool. The FNI extends to the surface of the pool and is filled with air. A shield plug consisting of polyethylene, borated graphite, and lead is used as a biological shield to prevent streaming to the pool surface. Lead and borated aluminum shielding on the perimeter of the tube provides shielding against gamma rays and thermal/epithermal neutrons, resulting in a flux with a greater fast neutron component. The shielding materials of the FNI are shown in Figure 1-8. Irradiations at the FNI are generally performed with the reactor near full power.

Figure 1-8: Fast Neutron Irradiator (“FNI”) geometry with materials labeled. Inner Diameter of the FNI is 10 inches.
Relative to the open pool position, the FNI is the experimental facility with the greatest negative reactivity worth. The negative reactivity worth of the FNI, combined with reactivity loses due to temperature, xenon, and fuel burnup can make it impossible to reach full power at the FNI, especially later in core life (i.e. approximately 2 years since the most recent refueling). In such cases, irradiations are performed at a lower power, with the irradiation time increased appropriately.

Section 1.3.2: The Fast Flux Tube

Like the FNI, the Fast Flux Tube (FFT) is an air-filled tube permanently installed in the north end of the pool. The FFT has an inner diameter of 6.25 inches [6]. The radius and shielding thickness of the FFT are both less than those of the FNI. Unlike the FNI, the shielding used in the FFT is cylindrical, introducing a significant water gap between the face of the reactor core and the shielding of the FFT. The reactivity effect of the FFT is significantly less than that of the FNI. Irradiations at the FFT are generally performed at low powers, often less than 1kW, and have a lower fluence target than runs performed in the FNI. Figure 1-9 shows a cutaway view of the shielding materials in the FFT.
Section 1.3.3: In Core Dry Tubes

Two aluminum dry irradiation tubes are permanently installed in the reactor core, shown in magenta in Figure 1-10. Each dry tube has an inner diameter of 3.175cm (1.25 inches) and a thickness of 0.159cm (1/16 inches). The dry tubes are installed near the front of the core in the outermost ring of fuel, placing them adjacent to 4 fuel elements and 2 water filled positions. Aluminum spacers are available to position samples at different heights along the axial direction of the core. To prevent streaming, each dry tube has two large radius bends, thus eliminating any
direct path to the surface. To mitigate activation of air, each tube has a rubber stopper, which prevents activated air from escaping the tube.

The position of the dry tubes with a core loading has limited flexibility. The dry tubes have historically been located in the E4 and E13 grid plate positions so that the vertical section of the dry tubes above the large radius bend is flush with the structure of the reactor tower. Changes to the structural supports of the upper grid plate as part of the new core moderator assembly project may warrant construction of new dry tubes to keep access to the core region clear for fuel handling and experiment setup. The change in position of the dry tubes is shown in Figure 1-10.

![Figure 1-10: Change in dry tube positions shown on Core 58A map](image-url)
Section 1.3.4: The Central Thimble

Located at the center of the core, the central thimble is a water filled aluminum tube of inner diameter 1.33 inches and outer diameter 1.5 inches having a peak thermal flux of approximately $3 \times 10^{13}$ n/cm$^2$*s. The tube extends below the core to the safety plate, and above the core past the surface of the pool. A cutout in the side of the tube several feet below the surface allows the central thimble to remain filled with water. This cutout also allows samples to be inserted and removed from the central thimble without removing them from the pool, which provides significant shielding. [1]

Section 1.3.5: Ex-Core Irradiation Positions

The 71,000-gallon pool in which the reactor is housed provides not only shielding, but also significant flexibility to relocate experimental equipment. With the reactor in the open pool, approved experiments can be freely positioned at various positions relative to the core. Small samples or detectors can be positioned directly on the front of the reactor’s lower grid plate and aligned using two rows of alignment holes similar to those used to position fuel. Larger samples can be maneuvered within the pool using a tower suspended from an instrument bridge, giving a full range of motion independent of the reactor bridge/tower.

Section 1.3.6: The Pneumatic Transfer System

The pneumatic transfer system, more commonly referred to as the “Rabbit”, provides a means of rapidly sending a sample from the Radionuclide Applications Laboratory into the core region for irradiation and returning the sample to the laboratory. The irradiation terminus of the rabbit system can be placed in various positions both within the reactor core and surrounding experimental equipment. The terminus was previously installed in a port in the graphite box of the old D$_2$O tank, and is now located in an irradiation port of the new core moderator assembly.
If a higher flux is needed, the terminus can be temporarily relocated to an in core position, however while the rabbit is installed in-core, the core must remain in the open pool and can not be coupled to other experimental facilities. With the rabbit installed in the D$_2$O tank, movement of the reactor core is not restricted.

**Section 1.4- Description of the New Core Moderator Assembly**

A new core moderator assembly was designed and analyzed by Uçar in 2013 as part of his PhD dissertation. [5] The goal of this new design was to allow for the use of five new beamports, improve the neutron to gamma ratio within the beamports, and provide additional beamlab facilities such as a cold neutron source with multiple cold neutron beams. After investigating various shapes for the new moderator assembly design, Uçar determined that a crescent shaped D$_2$O tank, shown in Figure 1-11, provided the optimal configuration since it allowed multiple beamports to be coupled to the system while simultaneously eliminating the contamination from 2.2 MeV gamma rays due to neutron capture on hydrogen.

![Figure 1-11: New core moderator assembly and reactor core geometry model. No structural materials are included in this model (Uçar Figure 5-3 [5])](image-url)
After determining that a crescent shaped moderator would be utilized, Uçar performed analysis to show that the new design would not violate the Technical Specifications limit for excess reactivity. Uçar’s analysis of core parameters will be expanded upon in Chapter 5. Through an iterative process, the radius of the tank and distance from the core to each beamtube was varied in order to achieve the greatest ratio of thermal to fast neutron flux at the far end of each beamtube. Figure 1-12 shows a simplified version of the model used to determine optimum dimensions.

![Figure 1-12: Beamtube flux optimization model used by Uçar with New Beamports labeled (Uçar Figure 5-9)](image)

The new core moderator assembly includes the D$_2$O Tank and beamports, a support tower for the tank, a new reactor tower and grid plates, and various structural components, and was built with funding obtained from the DOE. Figure 1-13 shows two more detailed computer-generated images of the new D$_2$O tank design.
The PSBR was shut down on May 17, 2018 for the installation of the new core moderator assembly. Following de-fueling of the core, the old reactor tower was removed from the reactor bridge and stored in the north half of the pool. The pool was south end of the pool was isolated using the pool divider gate and drained in accordance with facility procedures on May 24. After allowing the south pool to dry over a long weekend, RSEC staff and interns worked in shifts to set up shielding and remove the old D₂O tank, shown in Figure 1-14, while minimizing worker dose. Even after several weeks of decay time, the exposure rate at the face of the old D₂O tank was several R/Hr.
With the old D₂O tank and all other radioactive material removed from the south pool, the next step was to modify the pool wall and biological shield for the new beamports. The seven existing beamports were removed intact through a core drilling process. All material removed was surveyed for activation or contamination before removal from the reactor bay or neutron beam laboratory. Figure 1-15 shows a view of the south pool wall during the removal of the old beamports and drilling of the new beamport penetrations. As can be seen, the varied heights of the old beamports complicated this process, sometimes requiring overlapping wall penetrations such as that for the Cold Neutron Beamport replacing Beamport 2.
The final beamport installation from the neutron beam lab side is shown in Figure 1-16.
Each beamport, with the exception of the CNBP, has a cutout in the biological shield in which a shutter will be installed. The Cold Neutron Beamport contains a single round beamtube which will contain multiple neutron guides. The locations where old Beamports 1, 3, and 6 entered the beamlab are visible as circular patches of newly sealed concrete. The other old beamports are hidden by the new beamports and their shutters. Unlike the previous beamports, all of the new ports are at the same height.
Figure 1-17 shows the new D$_2$O tank in the south pool after installation. A new coat of sealant was applied to the entire south pool to patch the section that was removed to access the beamtubes as well as to repair any damage that may have occurred during the work in the pool. The south pool was re-filled on August 7, 2018. Loading and testing of Core 58A began shortly afterwards.

The initial startup procedures for Core 58A focused on verifying that no unexpected changes had occurred as a result of the tower replacement. One of the greatest concerns was positioning of the Core Instrumentation (Figure 1-3). The rack at the rear of the core which holds the detectors is adjustable on both the old and new towers. The positions of these holders on the new tower were set based on the observed positions of the originals, however the ability to accurately measure these positions was limited by the fact that the old tower was too radioactive
to remove from the pool. All observations were made with an underwater camera or from the surface of the pool 24 feet above.

When installing the Safety System Fission Chamber into its holder on the tower, an unexpected complication arose. Though the new detector holder was made to the exact specifications of the original, the detector exhibited binding as soon as it began to enter the holder, then became difficult to remove. The original holder was carefully removed from the tower and brought to the surface of the pool. It was found that while all dimensions of the old and new detector holders matched, the original had a small vent hole in the base to allow water to escape. Replacement holders for both fission chambers were manufactured and were installed without removing the tower from the pool. As a result, the measurements of exact detector positions made before installing the new tower are no longer valid for the fission chambers. The position of any detector included in a computer simulation will be based on observations made from the pool surface as opposed to exact measurements as a result.

Any uncertainty in reactor power as a result of detector position changes was accounted for by using a special startup procedure [7] during the first startup past the point of adding heat, then performing a thermal power calibration [3]. This special procedure performed step changes in power while comparing fuel temperature and reactivity loss data to measurements from Core 58. This data showed that the indicated power agreed well with historical temperatures, allowing the reactor power to be increased to a high enough power to perform a thermal power calibration, first in the open pool, then at the new D₂O tank. Figure 1-18 shows the reactor operating at the new D₂O tank during thermal power calibration.
Section 1.5- Fuel Management Before the New Core Moderator Assembly

Since 2006, fuel management at the PSBR has been performed using the TRIGSIMS code. TRIGSIMS was developed by Chanatip Tippayakul as part of his Ph.D. work [8]. TRIGSIMS couples the Monte-Carlo transport code MCNP [9] and SCALE, with MCNP being used to find power distribution and one-group reaction cross sections; SCALE, specifically ORIGEN-S, is used to determine compositions of burned fuel materials. In addition to providing the ability to perform burnup calculations, TRIGSIMS provides a front end for setting up core loadings and analyzing different configurations of control rod positions and fuel temperatures. TRIGSIMS has been used as the primary analysis tool for core loadings since Core Loading 53 in 2006 and remains an integral tool for the PSBR.

The TRIGSIMS code utilizes a predictor/corrector scheme to determine material compositions at the end of a burnup step. Burnup calculations are performed with all control
rods fully withdrawn from the core. Power distribution is found at the beginning of the time-step using materials specified in the input deck. This power distribution is used to estimate the effects of burnup upon materials. These burned material definitions are then used as an input for the corrector step to estimate the effect of material changes upon power distribution. An average of the two material definitions is then saved as the result of the burnup calculation as shown in Figure 1-19.

![Figure 1-19: TRIGSIMS burnup scheme flowchart [10]](image)

Each time that TRIGSIMS is run, power distribution and $k_{eff}$ data is calculated for both the predictor and corrector step and saved to an output file. For a core loading input deck, users must specify average reactor power, number of days at power, and temperature. The temperature can be independently defined for fuel, water, graphite, and zirconium, but is uniform in each
material. A “branch” card is also available to perform calculations that vary either temperature or control rod position. Any time that TRIGSIMS is run to calculate power distribution, the predictor/corrector steps will be performed, even for cases where only information at the current burnup is desired. This means that computer time will still be spent on these calculations, even if they will not be used within the final analysis report for the core loading.

For new core loadings prior to the installation of the new D\textsubscript{2}O tank, a total of 15 cases were analyzed for each core loading requiring 8 TRIGSIMS input decks to be manually prepared. Table 1-1 summarizes the cases analyzed for Core Loading 58, the last core loading to be used with the previous D\textsubscript{2}O tank. Each case is identified by two uppercase characters to indicate either that it is an established core loading, “CL”, or a potential new core loading “X<Y>” with the second character here indicating unique fuel patterns. This is followed by the number of the core loading being analyzed. Last is a lowercase letter indicating the conditions being analyzed. As an example, the XD58d case presented in Table 1-1 was an experimental loading (X). It was the fourth (D) unique pattern being considered for Core 58 and was set up to analyze the D\textsubscript{2}O tank at low power/temperature conditions (d).
The cases analyzed for a new core loading have evolved over time. The RSEC procedure for analysis of new core loadings, SOP-3 Core Loading and Fuel Handling [12], requires that a computer model be used to predict:

- Core Excess
- Location of the fuel element with MEPD
- Maximum normalized power
- Changes in peaking factors as core burnup occurs

The procedure also states that, if the code is capable, the following be predicted:

- Peak fuel temperature in the MEPD element
- Peak fuel temperature in the element with the maximum NP for the maximum allowed pulse.
- Flux peaking in water holes or voided positions within 4 inches of the center of the core.

The cases summarized in Table 1-1 address the four required predictions of SOP-3.

Additionally, the information provided by the TRIGSIMS code can be utilized with several...
empirical equations from the SAR in order to predict fuel temperatures at both steady state power and peak pulse temperature [1].

Several of the cases can be seen to be virtually identical to each other. An example of this can be seen in the XD58a.inp and XD58h.inp cases. The only difference between these two inputs is that the h-cases include a branch card to simulate additional control rod configurations. These redundant cases arose from the long simulation times involved when designing core loading with TRIGSIMS without running MCNP on multiple computer processors. The a-case simulation was needed to quickly estimate reactivity changes while the h-cases provided all of the same information in addition to effects of control rod position. Improvements in computer resources, specifically parallel computing capabilities, now allow all h-cases to be completed in less time than the single predictor/corrector for the a-case at the time TRIGSIMS was developed.

When applied to core loading cases using the new D$_2$O tank, the existing set of analysis cases face several challenges. Most importantly, the ability of TRIGSIMS to model a D$_2$O moderator tank is built around the assumption that the tank is cylindrical. Parameters such as radius, thickness and wall thickness can be specified. These parameters are inadequate to fully describe the new core moderator assembly. Figure 1-20 shows the geometry modeled by TRIGSIMS for a core loading in both the open pool and at the D$_2$O tank.
When discussing changes to core loading patterns, it is important to be able to accurately describe fuel positions within the reactor core. Multiple coordinate systems have been defined over the years for this purpose, each with its own merits. These systems will be introduced here so that they can be utilized freely to describe changes throughout this dissertation.

The first set of descriptors used to describe the reactor core are general directions and references to fixed core features. Though the core is roughly hexagonal, the side that faces north when in the open pool position is generally referred to as the “Core Face”. This side of the core is opposite the detectors, and is the most accessible, with the other four sides of the grid plate being surrounded by the reactor tower. The core face is used for coupling to experimental facilities and has a section of lower grid plate in front of it on which lightweight equipment can be placed. Space on the lower gridplate is also available on the sides of the core where long-term experiments have historically been located. With the installation of the new D$_2$O tank,
these core side positions will now be limited to a small space between the tank and the back three rows of the upper grid plate. When describing movement of fuel or objects within the core region, the term “forward” is often used to describe movement towards the core face. Figure 1-21 illustrates this core-face based reference frame. Another set of relative terms used to describe position within the core would be “in” and “out”, with “in” referring to nearer the central thimble and “out” meaning away from the central thimble.

![Diagram of Core Face, Front/Forwards, Back/Rear](image)

**Figure 1-21: Core position descriptions (adapted from SAR Figure 4-5 [1])**

The next set of descriptors used for describing core positions is a row/column coordinate system. The core is divided into 13 rows labeled A through M. With the reactor in the open pool position, Row A is at the north end of the lower grid plate and columns are numbered from west to east. In addition to providing an exact coordinate set that can be referenced in logs and procedures, this system makes identifying and communicating positions easy for fuel handling.
A downside to this coordinate system is that it is based on the lower gridplate, not the upper. This means that not all positions are available to place fuel or experiments. The core face is located at row C, with rows A and B having no fuel positions making them available only for locating experiments such as temporary dry tubes. Figure 1-22 illustrates this coordinate system.

The third method commonly used to describe positions within the core is by “Rings”. This system is based around the central thimble. The six fuel elements adjacent to the CT are referred to as the B-Ring. The twelve positions, including the Safety and Transient Rod locations, that surround the B-Ring make up the C-Ring. This naming scheme extends outwards.
as shown in Figure 1-23. The advantage of this system is that by describing proximity to the center of the core, it is useful in describing the impact of a position upon power distribution. For example, a new fuel element added to the B-Ring can be expected to have a significant power density while that same element added to the E-Ring would not. A disadvantage of this system is that it does not describe exact locations. At the periphery of the core, this system also begins to include positions that are not available for placement of fuel. The front and back of the G-Ring are located outside of the upper grid plate.

The next system used to identify positions within the core is a triangular lattice coordinate system. This system is used by both TRIGSIMS and MCNP to define the position of fuel elements, using the central thimble as the origin. Unlike standard cartesian coordinates, the y-axis is at a 45° angle to the x-axis, as shown in Figure 1-24. This allows all positions within
the lattice to be described by integer coordinates. Though commonly used in computer codes, this system is otherwise unfamiliar to staff at the PSBR, and is not used in procedures or logs.

![Triangular lattice coordinate system map](image)

*Figure 1-24: Triangular lattice coordinate system map [10]*
Chapter 2. Verification and Reconstruction of the Neutronics Model for the New Core Moderator Assembly

A detailed MCNP model of the new core moderator assembly is required for analysis not only of the new beamports, but also to predict effects of the new equipment on the reactor core. While the original model was adequate for optimization, a more detailed model of materials and geometry is needed to support analysis of future experiments. In the first section of this chapter, the original model, including its assumptions and limitations, will be discussed. Next, an updated model will be introduced with emphasis on ease of use and portability to future core loadings.

Section 2.1- Description of Previous Modeling Techniques

The original neutronics model of the new core moderator assembly created by Uçar consisted of multiple MCNP input decks which focused on subsystems of the new design. [5] Depending upon the analysis being performed at the time, varying levels of detail were included in a given simulation. Several simplifying assumptions were made throughout the design process which can easily be addressed now that a nearly finalized design is available. An overall view of the new core moderator assembly and beamtube model is shown in Figure 2-1. This figure is presented as a sketch since no complete MCNP model showing the entire system was available to create a geometry printout. The full geometry within each beamtube was included only in models of the beamtubes themselves, and not the entire system.
The first major assumption which went into the design of the new D$_2$O tank was to assume that the fuel materials present in the reactor core at the time of the work were representative of a “normal” core loading of the PSBR. All work performed by Uçar utilized the fuel materials for PSBR Core Loading 53G, shown in Figure 2-2, at the time the core was first loaded. The use of Core Loading 53G allowed the model to be compared with measurements performed using Beamport 4 of the existing beam facilities, shown in Figure 1-7. As can be clearly seen on the core loading map, Core 53G contained 10 graphite reflector elements along the periphery of the core. These elements were utilized as an added reflector in order to decrease leakage, and allowed the core to operate for an extended time without refueling. When modeling the new core moderator assembly, all 10 of the graphite reflectors were removed from the core. This allowed the new moderator to be positioned directly adjacent to the fuel to better couple.
with the core. The graphite reflector elements were removed from the core in 2010 during a routine fuel inspection, and are not expected to be utilized in future core loadings.

![Figure 2-2: PSBR Core Loading 53G][13]

The next assumption utilized in the design process was to fix the distance between the core and the moderator at 0.25” on all sides of the core where the moderator was present. This allowed for an easy comparison of potential geometries of well-coupled systems. This same clearance was utilized in the Computational Fluid Dynamics analysis performed by Uçar as the distance from the fuel to the surface of the D$_2$O tank, introducing a discrepancy between the thermal hydraulics and neutronics analyses. An extension of this assumption is that the structural materials of the moderator assembly, i.e. the tank itself, would have a negligible effect.
on the optimization problem. Since some form of structural material would be required in all cases, this assumption is valid. Neglecting structural materials of the tank designs, some of which had moderately complex geometries, made the process of setting up MCNP inputs significantly easier.

The last major assumption is that the beamtubes and the moderator can be modeled separately. This allowed the geometry to be split into multiple models that could be run independently. The complex geometry contained within each beamtube was simulated in such a way that the full core model did not need to be run each time to account for changes in geometry far from the core. This is a completely valid assumption and was necessary in order to accelerate the simulation process using the variance reduction techniques selected. These techniques will be discussed in Chapter 3.

Section 2.2- Requirements and Design Process for an Updated Model

When the design and assembly of the new core moderator and beamtubes are completed, a more detailed neutronics model is needed than is currently in use. This model must include not only the moderator material, but also any structural materials present in the design which could influence neutron flux in the core or beamports. The model also needs to be updated to use current fuel burnups. Since Core Loading 53G, there have been eight changes in the core loading of the PSBR, with Core Loading 58A currently installed. At a minimum, the new model must account for the current core loading, but it is preferred that the model be designed in such a way that it can be easily updated for any future core loading of the PSBR.

Another goal in designing an updated model of the core moderator assembly is to make the model as flexible as possible for any as yet unforeseen applications. Splitting the model into multiple subsystems to analyze each beamtube individually produced the desired results for
neutron flux and gamma intensity at the exit of the beamtubes, however it is incapable of predicting flux elsewhere in the system, such as at the pool wall (a value which is desired for analyzing new reactor operating positions. [14])

Development of the updated model started with the input deck for Core Loading 57 created by TRIGSIMs, the PSBR’s fuel management code. The geometry used in the TRIGSIMS model is limited to the core, a cylindrical D$_2$O tank, and water to a distance of 4 times the fuel pitch beyond the core or tank. The geometry was intentionally limited to ensure that time was not spent simulating particles that would not contribute to the power distribution within the core. For fuel burnup and power distribution calculations, this is a very useful simplification. When attempting to model larger experimental facilities such as the new core moderator assembly and beamports, this geometry must be significantly expanded. Figure 2-3 shows the geometry modeled by TRIGSIMS for Core Loading 57.
The geometry entries in the updated model were created based on input decks provided by Uçar [5] and design drawings available at the time. For each geometric feature added to the model, it was verified that both of these sources agreed. In cases where multiple geometries were utilized in Uçar’s work without one of them clearly being denoted as “final”, the one that was present in the engineering drawings was implemented. An example of this is the core side end of each beamtube. Within some inputs, such as that presented in Figure 1-12, the end of the beamtube was modeled as parallel to the core face. In other inputs, such as Figure 2-1, the end of each beamtube was perpendicular the axial direction of the tube. Multiple updates to the model have been necessary throughout the design and installation process, the most recent being
to incorporate as built measurements performed during the installation of the new tank and tower.

Figure 2-4: Preliminary model of the New Core Moderator Assembly including structural elements of the new D$_2$O tank used to predict reactivity and power distribution effects upon Core 57.

Figure 2-4 shows the earliest reconstruction of the core moderator assembly model. It can be seen that the geometry vastly exceeds that modeled by TRIGSIMS. At this stage of developing the model, emphasis was placed on defining the geometry in such a way that it can be easily modified in the future without significant changes to the input. The volume of D$_2$O within the tank is modeled as a single cell allowing for the material definition and properties to be easily changed. The walls of the tank itself are also modeled as a single cell but can easily be subdivided if needed. Each beamtube, including the air volume inside it, is modeled along the y-axis then translated to the correct orientation. This allows for collimators, shielding, or tally volumes to be added using only the axial dimension and a pre-defined translation card without
the need to calculate the exact coordinates of the new geometry. This early model was used for several simulations in order to gain a better understanding of the effects of the new moderator assembly on the current core loading, as discussed in Chapter 5.

Building upon the initial model, the geometry was expanded to include the pool wall, pool floor, and part of the Neutron Beam Laboratory (NBL). This model is shown in Figure 2-5. The water in the pool was modeled up to the surface of the pool. The Fast Neutron Irradiator tube was modeled in the appropriate position in the north end of the pool. Expanding the model to include the entire pool does slightly increase the computer time needed to run simulations due to greater distances at which particles must be tracked. Variance reduction techniques, which will be discussed in the next chapter, can be used to minimize this inefficiency by limiting the time spent tracking particles in the large volumes of the model unlikely to contribute to the flux at the ends of the beamports.

Including the entire pool and multiple experimental facilities in a single model has several benefits. Firstly, it eliminates the need to maintain a separate model for each experimental facility. In the past, this has led to simulations being performed using previous core loadings simply because the geometry of an experimental facility had not been transferred to the new core loading model. Manually adding geometry definitions to new input decks created by TRIGSIMS can be a time-consuming process and often suffers from numerous human errors as newly defined geometry must be copied and pasted to multiple locations within the TRIGSIMS generated input deck.
Figure 2-5: Early version of full pool model showing the reactor core coupled to the new D₂O Tank.

An additional benefit of this design is the flexibility to analyze scenarios that would previously have required significantly greater effort. The newly developed model allows the
user to translate and rotate the reactor core within the pool by modifying a single line of the input deck. Approved operating positions such as the R1 position and coupled to the FNI are already defined within the updated model.

Analyzing slight misalignments in reactor position is also possible with the newly developed model. For example, running a simulation with the reactor properly coupled, then repeating the simulation with the reactor translated a fraction of an inch away from the tank and comparing the tallies could simulate the effect of failing to properly couple the reactor to the new D$_2$O tank. In the previous model where the entire geometry was defined relative to the center of the core, making this very small change would have required modifying potentially hundreds of lines of input. The skewed case presented here is for illustrative purposes only, being suggested before the installation of the new hardware, and far exceeds the degree of misalignment that is possible for the new tower assembly. This flexibility within the model does however allow for the analysis of more realistic scenarios should they become apparent in the course of reactor operations or design of experiments.

Section 2.3- Features of the Final Updated PSBR Model

The final model produced as a part of this project greatly increases the ease of use when modeling experiments in the PSBR. Starting from the input deck created by TRIGSIMS, the geometry of the pool and all experimental facilities is automatically added using a script. This ensures that there is less possibility for human error in the process of defining geometry. Extensive testing performed during the development of the model has been done to eliminate any conflicts between the geometry and material definitions provided by TRIGSIMS and those used to model objects outside of the core. A key issue was to ensure that the core could be moved to any realistic position within the reactor pool without introducing geometry errors to the MCNP.
model. This required careful consideration of all geometry definitions for experimental facilities. Throughout the design and installation of the new moderator assembly, the model was continuously updated as new information became available. This includes updated dimensions of the tank, including an increase in thickness from 0.25” to 0.50” as a result of structural analysis, cold source geometry as specified by Eyers [15], and as built clearances between fuel and tank measured by PSBR staff during installation. The final model is shown in Figure 2-6.

Figure 2-6: Final full pool model including Core 58A
To further simplify the new model for future users, the script has been given the capability to automatically generate a separate model for each reactor operating position. This is done by applying a translation and rotation to the reactor core and restricting the bounding surface of the model. The input decks for each of these sub models still contain definitions for all geometry and materials of the full model, but provide a more focused view and can be more easily utilized with variance reduction techniques (described in Chapter 3). The geometry included in each sub model is shown in Figure 2-7 through Figure 2-9.

Figure 2-7: Tank sub-model of Core 58A
Figure 2-8: Open pool sub-model of Core 58A

Figure 2-9: FFT and FNI sub-models of Core 58A
Chapter 3. Variance Reduction Techniques Employed

The majority of the analysis of the new core moderator assembly is being performed using the MCNP6 code [16], which has been an integral part of the RSEC’s fuel management program for over a decade. Using MCNP for this analysis allows for repeatability in analyzing future cores since the loading pattern and material definitions must already be generated by TRIGSIMs when designing new core loadings. Because of the long streaming paths present in the new core moderator assembly, analogue transport methods are prohibitively slow for the simulation of experiments in the neutron beam laboratory.

In this chapter, after providing a benchmark case to show the need for variance reduction techniques, the methods previously used to model the new core moderator assembly will be discussed. A new variance reduction technique will be introduced which has the potential to simplify the process of simulating the new core moderator assembly, and preliminary data from this technique will be presented. Finally, the software tools being written to streamline this analysis will be discussed.

Section 3.1- Review of Previously Employed Variance Reduction Techniques

Surface sources have been used in the analysis beamtubes at the PSBR since Sarikaya et al. used the technique to predict flux spectra for Beamport 4 [17] and were also used extensively by Uçar [5]. The concept behind the surface source method is very simple. The problem geometry is divided into two parts; in this case the core region and the beamtubes. It is not necessary, and in fact inadvisable, to completely remove all geometry of the beamtubes from the core region model since neutrons can pass through the beamtube before scattering down its axial direction. With the geometry divided, the core region model is run and the neutron fluxes and
gamma intensities at the surface source are recorded. Since the surface source created by this model serves as the input for the beamtube model, it is important that the flux distribution at this surface is sufficiently detailed. For Monte Carlo based simulations, this can be a time-consuming process.

Several methods exist within MCNP to create a surface source. A built-in surface source generator, the SSW card, can be used to record all particles crossing the surface of interest and output them to a file. This ensures that the simulation remains analogue but limits subsequent simulations to running the same set of source particles. The output of the SSW card can be easily used as the source in additional runs, and can save considerable time, especially running variations on geometry far from the core region which have little to no impact on the reactor itself.

Another method of using surface sources is to define a tally on the surface source to record the flux spectrum, and then use the results of the tally to define a source distribution. Unlike the built-in surface source generator, this allows for a finely binned histogram or continuous distribution to be used as a source for subsequent simulations. While this may seem preferable to running a limited bank of recorded particles, it can be difficult to record adequately detailed tallies, which must include not only energy distribution, but also angular distribution. In highly anisotropic regions, such as the core side end of a beamtube, enough histories must be run to fully characterize the flux spectrum for each bin a histogram of angularly sampled bins.

Once a surface source is generated, it is still necessary to perform transport from the surface source to the region of interest. When simulating the beamtubes, only a small fraction of the particles recorded at the surface source are of interest, specifically those which are travelling along the axial direction.
Point detectors are a built-in tally type within MCNP that perform deterministic scattering and transport calculations at each scattering event. When using point detector tallies, whenever a particle scatters, the code determines the probability of the particle scattering in the direction of the point detector tally assuming isotropic scattering. A virtual particle with a weight equal to the weight of the original particle times the probability of scattering towards the detector is created, and the original particle’s weight is reduced to compensate. A deterministic calculation is performed to find the probability that the virtual particle could reach the point detector without additional scattering, and the virtual particle’s weight is further reduced by this probability before being recorded as a contribution to the tally. By performing this calculation at each scattering or absorption event, it is ensured that each neutron history contributes at least once to the tally. While this contribution often has a very low weight associated with it, it still results in a significant number of scores on the tally and thus low statistical uncertainty.

At first glance, point detector tallies may seem like the ideal solution for use in simulations of the beamtube exits. They produce tally results with low uncertainty even with a limited number of histories. Unfortunately, there are several significant drawbacks to this technique. The most readily apparent drawback is the increase in computational time from performing deterministic calculations throughout the simulation. Depending upon the number of point detectors used, the size of the model, and the cross sections of the materials, this can significantly slow down the execution of an MCNP run. [9] If all tallies within the run are point detectors, then this added time on deterministic transport calculations is likely not to be a serious issue. If, however a point detector tally is added to a simulation that includes surface or volumetric tallies, the Figure of Merit (FOM) for these analog tallies will be negatively impacted.
False convergence is an additional concern when working with point detectors. Because every scattering event results in a contribution to the tally, point detectors quickly achieve low uncertainty, but only consider particles that have been tracked through analog simulations. Contributions from particles traveling down the beamtubes may be under-sampled or completely omitted from tally results if too few total particles are simulated. This effect has been observed when attempting to estimate dose rates in the Beamlab outside of the beamline. Until particles are simulated traveling down the beamtube, the tally results will provide a prediction of dose only for particles passing through the wall of the pool.

Another weakness of point detectors is the use of an isotropic scattering assumption. In systems where molecular scattering contributes significantly to the tally results, point detectors cannot produce accurate results. In Uçar’s work, this was mitigated by the use of surface sources [5]. The scattering from the new core moderator assembly was modeled in the initial run to generate the surface source, while the point detector modeled the transport through the beamtube. The only situation in this model that would be biased by the isotropic scattering assumption would be neutrons from the surface source which are scattered back into the tube by the surrounding water after leaking out (assumed to be a relatively small fraction of the total flux).

Section 3.2- Weight Window Variance Reduction Techniques

In order to predict the effects on beamtube exit fluxes from changes in core loading and experimental configurations, a new model has been developed which uses a different variance reduction technique than previously employed – the weight window method. First, the general weight window method will be introduced. Next, the built-in weight window generator and its
limitations when applied to this model will be discussed before introducing an external weight window generation code.

Weight windows function by superimposing a mesh over the geometry of a problem. Each cell of the mesh has a particle weight lower limit cutoff. Particles entering the cell with less than that lower limit is subjected to a roulette process. Based on the ratio of particle weight to cell weight cutoff, this process will either terminate the particle or increase its weight and continue tracking it. Conversely, if a particle’s weight when entering a region exceeds the weight cutoff by a user specified factor, the particle will be divided into multiple pseudo particles, which can be individually tracked, with a total weight equal to the original. To mitigate the creation of exceedingly large numbers of particles when entering regions of increasing importance, the maximum number of particles created during the splitting process can be specified by the user. This may result in particles exiting the splitting process with weights above the upper weight bound for that region. When averaged over a large number of particles, the total weight of particles that survive the roulette process is equal to the weight of all particles entering, thereby avoiding any bias to tallies calculated using these particles.
Section 3.2.1: Built in Weight Window Generator

MCNP has a built-in capability to generate a superimposed mesh of weight windows over a geometry using data from analog particle tracking. Because MCNP’s built in weight window generator relies on the results of tallies in a previous run, it is sometimes necessary to run a considerable number of histories in order to generate a weight window mesh. In problems such as the core moderator assembly, where only a very small fraction of the total neutron population reaches tallies at the end of the beamtubes, this can be highly problematic, and can also result in the generated weight window mesh not covering the entire geometry of the problem. Figure 3-2 shows an example of the weight window mesh generated by MCNP for a tally centered in NBP2.
To generate these weight windows, 12.5 million histories were run. As can be seen, non-zero weight windows, shown in color in Figure 3-2, cover only a very small fraction of the geometry. Variance reduction parameters have not been generated for the remainder of the geometry in this case.

To increase the fraction of the geometry for which weight windows are generated, it is necessary to increase the number of scattering events which contribute to the tally of interest. The most straightforward way of doing so is to run a greater number of histories, a computationally intensive endeavor which scales roughly linearly with computer time. Another means of increasing the number of scattering events which result in a score to the tally is to utilize a point detector tally, since all collisions result in a score to the tally. As long as the weight of the point detector particle is high enough, this causes every collision to contribute to
the weight window map. Figure 3-3 shows the results of MCNP’s built in weight window generator for a point detector tally in NBP2 after running 12.5 million histories. It can easily be seen that a significantly greater fraction of the geometry is covered by defined weight windows than when a standard (analog) tally was used.

![Figure 3-3: Results of MCNP’s built in weight window generator for a point detector tally in New Beamport 2](image)

Several interesting features of the weight window generator can be seen in Figure 3-3. Large regions of the geometry where the flux can be expected to be non-zero, specifically on the far side of the core from NBP2, do not show any contribution to the weight window generator. This is most likely a result of the probability of contributing to the tally being sufficiently low that the particles’ statistical weight fell below a cutoff and was not scored. Another feature that is visible within the figure is that there are several regions of the mesh that are defined but are completely discontinuous with the rest of the geometry. Regions in the wall surrounding the beamtube can be seen to have a lower weight window lower bound than other regions within the geometry but are disconnected because they are the result of single particles and are thus
significantly under sampled. The solution to this problem while using the built-in generator is to run more histories.

Section 3.2.2: ADVANTG Weight Window Generator

The ADVANTG (AutomateD VAriaNce reducTion Generator) code [18] greatly simplifies the generation of weight windows for MCNP problems. ADVANTG can be configured using a very simple input deck specifying settings, as well as an MCNP input deck specifying materials and geometry. There is no need to setup and maintain a separate model for use in the deterministic calculations, making this weight window generation technique far more flexible. The DENOVO code, which is included with the ADVANTG code package, is used by ADVANTG to calculate the flux distribution throughout the geometry. MCNP tallies can be specified in the input deck for ADVANTG as targets for weight window generation. DENOVO is used to calculate the adjoint flux distribution at the tally points. The product of the adjoint flux within each weight window cell is gives the importance of particles in that cell to the quantity of interest. Regions with a higher importance are then assigned lower weight window cutoffs. Unlike the test cases run using the built-in generator, ADANTG is capable of generating weight window cutoffs for the entire geometry of the problem.

Using a tally surface or volume near the end of a beamtube as an adjoint source, lower limits for weight windows for a particle P in phase space are calculated by ADVANTG as:

$$w_l(P) = \frac{2}{1 + r \phi^\dagger(P)} \frac{R}{\phi^\dagger(P)}$$

where r is a user specified ratio between weight window upper and lower bounds, R is the response of a tally integrated over all phase space variables, and $\phi^\dagger$ is the scaler adjoint flux. [18] [19]
Figure 3-4 shows the weight windows generated by the ADVANTG code for the full pool model for volumetric tallies located in each beamtube midway through the biological shield. In the figure, blue indicates a lower weight window cutoff, meaning particles will be preferentially transported towards these regions. Red/yellow indicates a higher weight window lower bound, meaning particles will be selectively terminated through a roulette process. While the entire geometry of the pool is modeled, the majority of the computational time within MCNP will be spent transporting particles with fractional weights through regions with low weight window lower bounds, minimizing the otherwise computationally intensive process of simulating neutron scattering through the large volume of water above the core region.
Figure 3-4: Results of ADVANTG weight window generator for tallies in all new beamports showing weight windows covering the entire geometry. Particles will be preferentially transported towards regions with a lower weight window cutoff.

Because a deterministic code, DENOVO, is used, it is necessary to divide the geometry into a mesh, thus losing some of the details of the geometry. Figure 3-5 shows the mesh used by
ADVANTG in the core and moderator region. It can be seen that some mesh cells, especially those in the core region and along the outside edges of the moderator assembly, contain multiple materials which will have their cross sections averaged. To achieve a more realistic prediction of flux distribution, and thus more efficient weight window definitions, both forward and adjoint, it is desirable to make the mesh as fine as possible. Fortunately, the accuracy of results from ADVANTG impact only the speed of the MCNP calculations, not their accuracy. Because of this, a tradeoff can be made in determining how finely the geometry is meshed in ADVANTG.

Figure 3-5: Coarse weight window mesh overlaid on core and moderator geometry

Applying the ADVANTG code to the geometry of the PSBR has faced several challenges. The MCNP model produced by TRIGSIMs is configured as a criticality problem, making it incompatible with weight window-based variance reduction techniques. The geometry of the core moderator assembly and beamtubes includes multiple streaming paths which are not
orthogonal and are not properly modeled using the default settings within ADVANTG. Finally, the spatial mesh of the weight windows must be aligned with the geometry and refined sufficiently to capture details of the streaming paths.

The first step in addressing these issues was to convert the MCNP input deck to a fixed source problem. Tallies were added to each cell of the fuel definition in order to determine the fission rate within that cell. The tallied fission rate from this first run was then used as the source definition of a fixed source model of the reactor. Within the fixed source model, the number of neutrons emitted per fission was decreased to zero using the NONU card, thus accounting for the fact that the fission source is already fully defined. This technique is very similar to a surface source but has several advantages. The tallies being used are in the region of the model with the highest flux and achieve low relative errors even with minimal computational time. Surface sources placed far away from the core require significantly greater numbers of histories to be run in order to achieve the desired uncertainties. Tallying on volumes in the core region also allows the tallies to be simplified to include less data. For a surface tally at the beamports, the flux recorded on the surface can contribute directly without scattering to the results of a tally in the final simulation. The energy, angle, position, and velocity of neutrons crossing the surface source must be recorded in sufficient detail as to not bias the results of subsequent tallies.

For a volumetric source in the core region, several assumptions can be made. As long as the fuel is divided into a sufficient number of regions, the source distribution within each fuel region can be assumed to be spatially uniform within each section. Additionally, since the fuel is predominantly $^{235}\text{U}$ undergoing thermal fission, a Watt fission spectrum can be assumed as the energy distribution of source particles. These assumptions drastically reduce the number of histories needed to fully define the volumetric fixed source. The error introduced by this
assumption can be easily quantified by comparing tally results outside of the core region with results from a kcode simulation, as will be presented for several cases in Chapter 4.

The next challenge faced when generating variance reduction parameters with ADVANTG was dealing with the streaming of neutrons and gamma rays down the beamtubes. Figure 3-6 shows the weight window lower bounds generated using default settings in ADVANTG for tallies in the beamtubes far away from the core.

DENovo utilizes the discrete ordinates method to solve the neutron transport equation [19]:

\[
\left[ \hat{\Omega} \cdot \nabla + \sigma(\vec{r}, E) \right] \psi(\vec{r}, \hat{\Omega}, E) = q_{ex}(\vec{r}, \hat{\Omega}, E) + \int dE' \int d\Omega' \sigma_s(\vec{r}, E' \rightarrow E, \hat{\Omega}' \cdot \hat{\Omega})(\vec{r}, \hat{\Omega}', E')
\]

Where:

\( \hat{\Omega} = \text{unit vector in direction of particle travel} \)

\( \sigma = \text{cross section} \)

\( \vec{r} = \text{position} \)

\( E = \text{particle energy} \)

\( q_{ex} = \text{an external source} \)

The first term of the transport equation, \( \left[ \hat{\Omega} \cdot \nabla \right] \psi(\vec{r}, \hat{\Omega}, E) \), defines the streaming of neutrons through a volume without interaction. Within the discrete ordinates calculations performed by DEVOVO, this term is evaluated for a finite number of directions, which may or may not align with the beamtubes. Even though the beamtubes present a direct streaming pathway between the D\(_2\)O tank and the region of interest, the solver may only be able to see a pathway between the point of interest and a scattering point at the wall of the beamtube.
The solution to this inability to identify streaming pathways within this simple core moderator assembly model was found to be changing the angular quadrature used by the DENOVO code from the default order of 10 up to an order of 20, the results of which are shown in Figure 3-7. This increased the number of angles per octant from 15 to 55, increased the probability that DENOVO was performing calculations along an axis roughly parallel to the orientation of each beamport.

Figure 3-6: Weight windows generated by ADVANTG code with default quadrature scheme (QR) of Order 20 showing no clearly identified streaming path along beamtubes.
With the quadrature order increased, it was next necessary to adjust the spatial mesh used by ADVANTG. The weight window meshes presented in Figure 3-6 are too coarse to clearly divide the beamtubes from the surrounding water. Decreasing the size of each special mesh to 1cm x 1cm x 1cm allows the transport simulation to clearly identify streaming paths, even along NBP4 which has a radius of 5.1cm and is not orthogonal to the mesh. Figure 3-7 shows a more finely meshed version of the higher quadrature order simulation performed in Figure 3-6. A region of higher importance (dark blue) can clearly be seen around each beamtube extending back into the moderator tank. This higher importance region was not found by DENOVO using the coarser mesh. When applying this refined mesh to the full reactor pool model, only the core,
moderator, and beamtube regions need to be finely meshed. A coarse mesh can be safely used in regions of low flux which are not likely to contribute to the tallies of interest.

To test the effects of the weight windows generated by ADVANTG, an MCNP model of the entire pool was configured with tally volumes near the ends of the beamtubes. The simulation was run both with and without weight windows, using the same number of histories in each run. To compare how quickly results were obtained for each tally, the Figure Of Merit (FOM) was used as a metric. The MCNP Manual defines FOM as:

\[
FOM = \frac{1}{\sigma^2 \times T}
\]

where \(\sigma\) = relative error and \(T\) = computer time in minutes. [16] This definition works well when comparing multiple simulations run on the same computer hardware when changes in parallel efficiency are negligible.

The results of these tallies are presented in Table 3-1. As can be seen, the tallies performed using weight windows had significantly lower errors.

Table 3-2 summarizes the ratio of the errors and the ratio of the figures of merit for each tally. Utilizing weight windows reduced the maximum relative error on the tallies to 25% of the error without variance reduction. At the same time, the figure of merit ratios indicate that tally results were achieved up to 84 times faster by using weigh windows. Note that the high uncertainties lead to low confidence in the exact value of the FOM. As such, all speedup factors presented here are approximate. These preliminary results are presented “as is”. Results presented for later tests will include lower uncertainties, as well as more detailed information on runtime.
Table 3-1: Summary of preliminary results for full pool model simulations

<table>
<thead>
<tr>
<th>TALLY</th>
<th>STANDARD</th>
<th>WEIGHT WINDOWS</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mean</td>
<td>Uncertainty(%)</td>
</tr>
<tr>
<td>NBP1</td>
<td>3.11E-08</td>
<td>42.99</td>
</tr>
<tr>
<td>CNBP</td>
<td>1.39E-08</td>
<td>45.61</td>
</tr>
<tr>
<td>NBP2</td>
<td>8.24E-08</td>
<td>30.95</td>
</tr>
<tr>
<td>NBP3</td>
<td>6.17E-08</td>
<td>30.96</td>
</tr>
<tr>
<td>NBP4</td>
<td>8.95E-09</td>
<td>100</td>
</tr>
</tbody>
</table>

Table 3-2: Ratio of errors and FOM for preliminary full pool model simulations (Weight Windows/Standard)

<table>
<thead>
<tr>
<th>TALLY</th>
<th>RATIO UNCERTAINTY</th>
<th>RATIO OF RUNTIMES</th>
<th>RATIO FOM</th>
</tr>
</thead>
<tbody>
<tr>
<td>NBP1</td>
<td>0.022</td>
<td>62.51</td>
<td>32.8</td>
</tr>
<tr>
<td>CNBP</td>
<td>0.138</td>
<td>0.62</td>
<td>84.3</td>
</tr>
<tr>
<td>NBP2</td>
<td>0.258</td>
<td>0.61</td>
<td>24.5</td>
</tr>
<tr>
<td>NBP3</td>
<td>0.231</td>
<td>0.61</td>
<td>30.9</td>
</tr>
<tr>
<td>NBP4</td>
<td>0.160</td>
<td>0.64</td>
<td>60.9</td>
</tr>
</tbody>
</table>

Section 3.3: Testing of Variance Reduction Parameters to Generate a Final Weight Window Library

Having shown that the ADVANTG code was capable of generating variance reduction parameters which outperformed the build in weight window generator, an effort was undertaken to generate a set of weight window libraries which could be used for future core loadings and to
identify optimal parameters for their generation. To allow multiple cases to be quickly analyzed, several restrictions were placed on the test cases:

- A single beamport was analyzed in each simulation. While ADVANTG is capable of generating weight windows for multiple points simultaneously, it is unlikely that more than one beamport at a time will be simulated for future experiments.
- A surface tally on the far side of the biological shield with no energy binning was used.
- All tests within a series were performed on the same computer using the same number of processors.
- FOM was calculated using wall clock time, not processor time. This includes any losses in parallel efficiency resulting from variance reduction in the FOM.
- The same spatial mesh was employed for the deterministic calculation for all cases involving the new core-moderator assembly.
- A baseline simulation for each beamport was performed using one hundred times the number of histories and no variance reduction.

Section 3.3.1: Use of ADVANTG’s Built in Quadrature Sets

Utilizing a higher order quadrature from ADVANTG’s built in libraries had already proven effective in allowing the code to calculate adjoint fluxes that account for the streaming paths introduced by the beamports, so this was chosen as the first parameter to investigate further. For these cases, the default quadrature scheme, Quadruple Range, was used. This quadrature has been shown to be effective for problems with streaming along the coordinate axes [20] (such as NBP2). A test script was written in python to repeatedly run the ADVANTG code with varying quadrature orders to generate weight windows, then run MCNP. Tally results were
automatically extracted and added to a custom output deck along with runtime information. Output decks from MCNP, as well as weight window files and silo files, which contain detailed adjoint flux distribution information, from ADVANTG were preserved for additional analysis. Results of this testing are presented in Figure 3-8.

Figure 3-8: Results of varying quadrature order on speedup when generating weight windows

Testing of quadrature order vs. speedup was first conducted using NBP2 and starting from quadrature order 8, the lowest order supported, and proceeding to the highest supported quadrature order, 32. During this sweep of quadrature orders, several observations were made...
that resulted in restrictions to the cases tested afterwards. Quadrature order 26 resulted almost immediately in the ADVANTG code crashing. Inadequate information was provided by runfiles to find and correct the cause of this crash. No weight windows were generated for this case and it was not included in the results. Quadrature order 8 resulted in a set of weight windows that caused MCNP to run much slower than the baseline case. This simulation was terminated after several days. For reference the baseline case with ten times as many histories completed in 14 hours. Testing of NBP3 and NBP4 weight window sets was conducted from highest quadrature order to lowest and was terminated when a single case was observed to run for more than a day. This restricts the data for NB3 to quadrature orders of greater than 14 and greater than 22 for NBP4.

Compared to previous simulations, the speedup factors observed are lower in these tests. This is believed to be a result of using a surface tally at the far end of the beam as opposed to a volumetric tally within the biological shield wall as the adjoint source. The newer methodology reduces the contribution of scattering from the walls of the beamtubes or pool wall to the figure of merit, making the problem more focused on streaming. These results also show that a single quadrature order was not found to be optimal for all beamports. Visual inspection of the adjoint flux maps produced by ADVANTG provide a clue as to the likely cause of this.

Figure 3-9 shows the adjoint flux for NBP2 and NBP3 calculated using a quadrature order of 30. While the adjoint flux along NBP2 increases gradually along the length and uniformly along the radius of the beamtube, the adjoint increases more rapidly at the center of NBP3 and spreads out towards the perimeter of the beamtube as the distance from the core increases. While this could simply indicate that neutrons at the center of the beam are more likely to reach the end, it could also be a sign that the quadrature being used does not line up
very well with the angle of the beamport. Ray-effects are also visible in along the quadrature
directions for the NBP3 adjoint flux.

Figure 3-9: Adjoint flux maps for NBP2 (top) and NBP3 (bottom) generated by ADVANTG using Quadrature Order 30
Section 3.3.2: Testing of Custom Quadrature Sets

Having tested multiple built in quadrature sets for use in describing the new beamports, it was found that no single set was ideal. To provide a greater level of control over how ADVANTG solves for forward and adjoint fluxes it was the decision was made to investigate manually specifying a quadrature set. Within a set, each vector is specified as a set of three cosines, one for each of the x, y, and z axes, denoted as $\mu$, $\eta$, and $\xi$ in Figure 3-10, as well as a weight. The full set of ordinates and weights in a quadrature set describe the directions in which angular flux is calculated, with the weight describing the portion of the unit sphere described by the ordinate. While the ADVANTG code uses the Discrete Ordinates method to determine the angular flux within each mesh cell, to save memory data is only saved for the scalar flux calculated as:

$$\phi = \sum_{m=1}^{M} \psi(\Omega_m)w_m$$

To determine the importance of aligning the ordinates of the quadrature set with each beamport and to test the ability of DENOVO to read external quadrature sets, a custom quadrature set was defined with an ordinate parallel to the beamport being tested and its rotationally symmetric equivalents on other octants. This test, while lacking in mathematical rigor, provided speedups that were equal to or better than those achieved using the built-in quadrature sets. Having demonstrated that this effect was significant to the speedup factor, two options were available: either the model could be rotated to work better with a built-in quadrature set or a quadrature set could be developed to suit the geometry. In the interests of ease of use for future users of the model, rotating the geometry by a different angle depending on the beamport was dismissed as impractical and confusing.
Ordinates and weights for several Legendre-Chebyshev quadrature sets generated by Walters [22] were provided for use in this analysis. The Legendre-Chebyshev $S_6$ quadrature set was selected for use in more detailed testing. With 15 angles per octant, multiple simulations using this quadrature set were able to be run in a reasonable amount of computer time (hours per beamport as opposed to days). Starting from this quadrature set, a new quadrature was defined for each beamport by adding a 16th ordinate that was parallel to the beamport in the $x$ and $y$ directions and slightly above the beamport in the $z$ direction (a vector perfectly parallel to the beamport would fall on the boundary between octants and is not permitted by the solver). The weight of the ordinate added was set based on the solid angle defined from by the exit area of the beamtube relative to a point at the entry of the beamtube. The weights of the other ordinates
were reduced accordingly such that the sum of all ordinates remained unity. Table 3-3 shows the weights of the ordinates in the original and modified quadrature sets. The high precision of the numbers specified here is necessary when providing custom quadrature sets to DENOVO.

Table 3-3: Original and modified weights for NBP3 quadrature set

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<th>Ordinate Number</th>
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<th>Modified Weight</th>
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<tr>
<td>0.0591048000</td>
<td>0.05909673136</td>
<td></td>
</tr>
<tr>
<td>0.0591048000</td>
<td>0.05909673136</td>
<td>0.00013651409</td>
</tr>
</tbody>
</table>

Figure 3-11: Single octant of quadrature for original quadrature set (left) and NBP3 custom quadrature set showing added vector in red (right).
To verify that the addition of a beamport parallel vector caused the new quadrature set to outperform the original when generating weight windows, the ADVANTG code was run for a tally at the end of NBP1 using both quadrature sets. Visual comparison of the adjoint flux calculated using the different quadratures clearly shows that the unmodified quadrature was not able to identify the streaming path created by the beamport as shown in Figure 3-12. Based on this adjoint map, no significant amount of particle splitting will occur to drive low weight particles into the beamtube, but if a neutron does begin to travel along the beamtube, it will enter a region where its estimated importance increases rapidly. Between where the beamtube leaves the D$_2$O tank and the tally volume, the adjoint flux increases by 11 orders of magnitude, introducing the potential for a single neutron to be split into $10^{10}$ pseudo particles, each of which must be tracked on a single processor.

Figure 3-12: Adjoint flux for NBP1 generated by ADVANTG using the unmodified Legendre-Chebyshev $S_6$ quadrature
Compared to the adjoint flux found using the unmodified quadrature, the solution found using the custom quadrature set clearly identifies the streaming path of NBP1 as shown in Figure 3-13. The increase in importance along the beamtube is only 6 orders of magnitude and particles will likely have undergone considerable splitting before entering this streaming path. This serves both to limit the computational time spent on particles travelling down the beamport by splitting less along its axis and to increase the total number of particles entering the beam by splitting within the moderator.

Figure 3-13: Adjoint flux for NBP1 generated by ADVANTG using the custom quadrature set
When MCNP was run using the weight windows created using these adjoint fluxes, the custom quadrature set run completed in under an hour with a FOM 32.2 times higher than the non-variance reduction baseline. The simulation performed using weight windows from the unmodified quadrature set ran for over a month without completing. Because no final estimate of error was produced, a FOM cannot be calculated. The best assessment that can be made of this result is that the addition of a beamtube parallel ordinate makes the variance generated weight windows usable when compared to the original quadrature set. Testing of the unmodified quadrature set on the other beamports was not performed due to the impractical amounts of computer time consumed during the simulation of NBP1.

In the absence of a reference solution to compare the flux predictions generated by the custom quadrature set, an alternate method was needed to evaluate the reliability of the quadrature. An even-moment test was applied to the custom quadrature sets to evaluate the quadrature set’s ability to accurately integrate a function, in this case the neutron flux, over the unit sphere. [23] For even moments, of the quadrature:

\[
\sum_{m=1}^{M} w_m \mu_m^n = \sum_{m=1}^{M} w_m \eta_m^n = \sum_{m=1}^{M} w_m \xi_m^n = \frac{1}{1 + n} \text{ for even values of } n
\]

The odd moments of the flux are automatically conserved since all the quadratures are rotationally symmetric. The results of applying the even test to the custom quadrature set for NBP1 are presented in Table 3-4. It can be seen that even for higher moment orders, the custom quadrature set produces a result nearly identical to the exact solution. The maximum observed deviation was for the cosine in to the z-axis for a moment of order 30 where the custom quadrature set differed from the exact solution by 0.52%. Because the flux calculated using this
quadrature is needed only as a means of accelerating a Monte-Carlo calculation, this deviation can be considered acceptable.

Table 3-4: Results of Even Moment Condition Test for NBP1 quadrature based on Legendre-Chebyshev S6 quadrature

<table>
<thead>
<tr>
<th>Even Moment Order</th>
<th>Exact: $\frac{1}{1+n}$</th>
<th>$\sum_{m=1}^{M} W_m \mu_m$</th>
<th>$\sum_{m=1}^{M} W_m \eta_m$</th>
<th>$\sum_{m=1}^{M} W_m \xi_m$</th>
<th>Average Difference [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>0.3333333333</td>
<td>0.33336895</td>
<td>0.33334309</td>
<td>0.33328797</td>
<td>0.000</td>
</tr>
<tr>
<td>4</td>
<td>0.2000000000</td>
<td>0.19999880</td>
<td>0.19997294</td>
<td>0.19997274</td>
<td>-0.009</td>
</tr>
<tr>
<td>6</td>
<td>0.14285714</td>
<td>0.14286459</td>
<td>0.14284496</td>
<td>0.14283763</td>
<td>-0.006</td>
</tr>
<tr>
<td>8</td>
<td>0.1111111111</td>
<td>0.11110819</td>
<td>0.11109480</td>
<td>0.11109590</td>
<td>-0.010</td>
</tr>
<tr>
<td>10</td>
<td>0.0909090909</td>
<td>0.09090387</td>
<td>0.09089520</td>
<td>0.09089662</td>
<td>-0.012</td>
</tr>
<tr>
<td>12</td>
<td>0.07692308</td>
<td>0.07691480</td>
<td>0.07690936</td>
<td>0.07691250</td>
<td>-0.014</td>
</tr>
<tr>
<td>14</td>
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<td>0.03438057</td>
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<td>0.03211700</td>
<td>0.03211694</td>
<td>0.03209102</td>
<td>-0.464</td>
</tr>
</tbody>
</table>

With these newly defined custom quadrature sets, additional tests were performed to see if the speedups from adjoint based weight windows could be further increased. Changes were made to the MXSPLN entry within MCNP to control the maximum number of splits per event. The default setting in ADVANTG is a maximum of 100 splits per event. Figure 3-14 presents the results of this analysis. Testing was performed from 25% of the default up to 200%.
For all beamports analyzed, the speedup factor showed a general increase up to a MXSPLN of 100, after which oscillations were observed. Under ideal circumstances, the speedup factor should increase asymptotically until the maximum number of splits permitted no longer exceeds that specified by the adjoint flux distribution. During these simulations, processor usage was monitored carefully. At the end of each run, the number of processors being used dropped from 20 (the total number used), down to several, and then down to 1 before the simulation eventually finished. The amount of time spent running on fewer than the total
number of processors was on the order of hours for the simulations with a MXSPLN greater than the default value and is expected to be the cause of the oscillations observed at high MXSPLN numbers. This lack of parallel efficiency results from a single starting particle being split numerous times as it enters regions with a low weight cutoff, making it more likely to contribute to the tally. [24] [25] When splitting particles, each source particle is assigned to a single process and cannot benefit from multicore computers. This trailing end of processor usage can be limited by either restricting the maximum number of splits, or by increasing the total number of particles such that the fraction of total runtime spent on the last few processors is minimized. For most cases when using the generated weight window libraries for simulations of the PSBR, the default MXSPLN value should be used.

Section 3.4: Generation of a Final Set of Weight Window Libraries for the New Neutron Beamports

A final set of weight windows was generated using the parameters identified previously. A custom quadrature was defined for each beamtube, and ADVANTG was run for each case. Figure 3-15 shows the adjoint flux for a tally at the end of NBP2 as calculated by DENOVO using the custom quadrature set. A larger version of this plot, as well as plots of adjoint flux for the other beamports, is available in Appendix B.1: Adjoint Flux Maps.
The resulting weight windows from this calculation are shown in Figure 3-16 with plots for other beamports available in Appendix B.3: Weight Window Map Sets. Each of these five weight window files contains 850 MB worth of data. In addition to using an extra 3.4GB of storage beyond the 12GB needed for cross section libraries used by TRIGSIMS, this makes transferring the weight window definitions to other computers problematic. The weight window definitions are too large to be transferred as an attachment on currently available email services and must instead be hosted on a remote server or transferred via removable storage. To mitigate the large file sizes, a feature of ADVANTG, mcnp_wwCollapse_factor, was utilized.
The weight window collapsing methodology works by calculating the forward flux, shown in Figure 3-17 in addition to the adjoint flux. The product of the forward and adjoint, the contribution flux or response density,

\[ C(\vec{r}, E) = \phi(\vec{r}, E)\phi^\dagger(\vec{r}, E) \]

shown in Figure 3-18, is then calculated for each cell. While the adjoint specifies the probability that a neutron at a given point will reach the point of interest and the forward flux specifies where neutrons are within the system, the response density gives information on where the particles reaching a point of interest are coming from [26] [27].

Figure 3-16: Final weight window set for NBP2
Figure 3-17: Forward flux calculated by DENOVO

Figure 3-18: Response density for NBP2 calculated by DENOVO
Having information on where the neutrons reaching the region of interest come from (and where they do not) allows the weight window definitions in regions of low response density to be merged. A collapsed set of weight windows was generated using an mcnpww_colapse_factor of 10. Each of these files is 85MB, making them significantly more manageable than the originals. Figure 3-19 shows the collapsed weight window set for NBP2. Several methods of collapsing weight windows are detailed by Ibrahim et al. [28]. The method utilized by ADVANTG, Weight Window Coarsening, identifies a slice of the volume integrated contribution flux consisting of a single mesh cell in the x, y, or z dimension and spanning the geometry in the other dimensions. The block with the lowest integral contribution flux is merged with a neighboring block by averaging forward and adjoint fluxes. This process is repeated until the total number of mesh cells is reduced by a user specified ratio.

Figure 3-19: Collapsed weight windows for NBP2 generated by ADVANTG
To verify the effectiveness of the newly generated sets of weight windows, a reference case for each model was run without weight windows using 100 times as many histories in order to limit the relative error. The runtime, tally mean, and COV were recorded for use in calculating speedup. Runs performed using weight windows utilized 12.5 million histories, the same number of starting particles normally used during core analysis and fuel burnup simulations. When utilizing a variance reduction technique, it is important to avoid biasing the mean. Figure 3-20 compares the means of the reference calculation with those of both full and collapsed weight window runs. Error bars shown are one standard deviation.

For NBP1 and NPB2, both weight window runs fell within one standard deviation of the mean of the reference case. For NBP3 and NBP4, the weight window runs were slightly outside of one standard deviation from the reference case. Considering the relatively high uncertainty on the reference cases, this agreement with the mean can be considered acceptable, especially considering that any experiments designed using this model will still need to be experimentally verified.
In the case of the Cold Neutron Beamport, the weight window simulations did not show good agreement with the reference case. There are several possible explanations for this disagreement beyond statistical error (the most likely cause). Unlike the other beamports, the CNBP is modeled as containing a vacuum instead of air. The CNBP is also much larger than the other beamports and increases in diameter as the distance from the core increases. The increasing diameter of the beamport is not likely to be well described by the single added vector in the custom quadrature set.
The inability of the weight windows to describe the CNBP is not as much of a setback as it may initially appear. This beamport is not intended to be utilized as a single open beam but will instead contain three curved cold neutron guides. The curved shape of these guides is likely to be impossible to simulate with any low quadrature order deterministic calculation. Additionally, even in the event a working set of weight windows could be developed for the CNBP, MCNP is not capable of correctly modeling the scattering of cold neutrons within the guides. Scattering of cold neutrons is heavily dependent upon the lattice structure of the guide, and cannot be simulated by the free gas thermal treatment and \( S(\alpha,\beta) \) correction used by MCNP for molecular scattering.

![Figure 3-21: Figure of Merit for reference, weight window, and collapsed weight window cases](image-url)
The net quantities of interest are the FOM and the speedup factor. Figure 3-21 compares the figure of merit for all cases. With the exception of the CNBP cases, several observations can be made for all beamports. The weight window and collapsed weight window cases all had higher FOM values than their reference cases. This indicates that the weight windows resulted in an increase in the speed of the simulation. The FOM of both the weight window and collapsed weight window cases were comparable for each beamport, showing little difference in speedup despite the significantly smaller weight window files.

Figure 3-22 and Table 3-5 show the speedup factor for each case. With the exception of the CNBP which showed no significant speedup, NBP3 exhibited the lowest increase in speed, running 17.3 times faster than the reference case. NBP2 showed the greatest speedup at a factor of 67.2.

<table>
<thead>
<tr>
<th>BEAMPORT</th>
<th>WEIGHT WINDOWS</th>
<th>COLLAPSED WEIGHT WINDOWS</th>
</tr>
</thead>
<tbody>
<tr>
<td>NBP1</td>
<td>32.2</td>
<td>30.0</td>
</tr>
<tr>
<td>CNBP</td>
<td>1.7</td>
<td>1.0</td>
</tr>
<tr>
<td>NBP2</td>
<td>67.2</td>
<td>66.7</td>
</tr>
<tr>
<td>NBP3</td>
<td>17.3</td>
<td>17.3</td>
</tr>
<tr>
<td>NBP4</td>
<td>58.9</td>
<td>56.1</td>
</tr>
</tbody>
</table>
In addition to the weight window libraries created using the modified Legendre-Chebyshev $S_6$ quadrature, testing was also performed using the same modifications to an $S_{30}$ quadrature. Unlike the lower quadrature order, the weight window libraries produced by the unmodified Legendre-Chebyshev $S_{30}$ quadrature were able to be run without any major difficulties. The modified version of the higher order quadrature set generally produced the highest observed speedups of all tested cases, and the weight window libraries prepared for use in future simulations have been updated accordingly. Figure 3-23 compares the speedup factors from these simulations with those for several previous calculations.
Section 3.5 Weight Window Libraries for the FNI and FFT

Weight window libraries were generated for the FNI and FFT. Unlike the new beamtubes, preliminary testing showed that default settings for ADVANTG were effective in achieving large speedups. This was expected because these experimental facilities are surrounded by water and do not involve long streaming paths. The speedup factors for both FNI and FFT were greater than 100. When combined with the higher fluxes resulting from being nearer to the core than the beamports, simulations of these experimental facilities are far simpler than the neutron beams. Tally results, FOM, and speedup data for the FNI and FFT simulations are presented in Table 3-6. Adjoint flux maps are available in Appendix B: Additional Figures.
### Table 3-6: Tally results and speedups for FNI and FFT

<table>
<thead>
<tr>
<th>CASE</th>
<th>TALLY</th>
<th>COV</th>
<th>RUNTIME [s]</th>
<th>FOM</th>
<th>SPEEDUP</th>
</tr>
</thead>
<tbody>
<tr>
<td>FFT KCODE</td>
<td>1.65E-05</td>
<td>0.0092</td>
<td>1.23E+04</td>
<td>0.96</td>
<td></td>
</tr>
<tr>
<td>FFT WEIGHT</td>
<td>1.70E-05</td>
<td>0.0019</td>
<td>2.50E+03</td>
<td>110.71</td>
<td>115</td>
</tr>
<tr>
<td>WINDOWS</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FNI KCODE</td>
<td>1.06E-05</td>
<td>0.0106</td>
<td>9.88E+03</td>
<td>0.90</td>
<td></td>
</tr>
<tr>
<td>FNI WEIGHT</td>
<td>1.07E-05</td>
<td>0.0022</td>
<td>1.79E+03</td>
<td>115.51</td>
<td>128</td>
</tr>
<tr>
<td>WINDOWS</td>
<td></td>
<td></td>
<td></td>
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<td></td>
</tr>
</tbody>
</table>
Chapter 4. New Software Tools for Modeling of the New D₂O Tank and Streamline Analysis of New Core Loadings

Throughout the process of updating the PSBR’s fuel management model, numerous scripts and tools were written to make the design and analysis process easier and less prone to human error. Even before the addition of the new D₂O tank, modeling the PSBR using TRIGSIMS was a complex process coupling multiple codes, each with their own input and output formats. Most of these tools were written to solve a single problem and are capable of being used independent of the others. In this chapter, each of the software tools developed will be described in greater detail to provide insight into how they function independently and how they have been combined into a single analysis package for new core loadings of the PSBR. Once all of the software has been introduced, a plan for its use in the design and analysis of Core Loading 59 will be described.

Section 4.1: Subroutines Prepared to Automate Existing Core Analysis Requirements

Section 4.1.1: Post_Trigsims

An executable titled “post_trigsims.exe” was provided with the TRIGSIMS code and its use documented in an appendix of the TRIGSIMS User’s Manual [10]. The purpose of this code is to merge the burned material definitions created by TRIGSIMS with a user provided input deck specifying fuel positions for a new core loading to create a TRIGSIMS input deck for a newly specified loading. This original post_trigsims code has not been successfully run by the PSBR staff since receiving the TRIGSIMS package. Because only an executable and no source code was provided for this tool, debugging the code was not possible. Until the design of Core
Loading 56, TRIGSIMS input decks for burned core loadings were created manually by copying and pasting data from TRIGSIMS’s runfile and previous input deck.

Shortly before Core Loading 56 was designed, a replacement for post_trigsims was written in python. Though it shares the same name and purpose, this script is completely independent of the original. This new script takes the input, output, and run files from TRIGSIMS and creates a new TRIGSIMS input deck with fuel materials burned as specified in the original input deck. The loading pattern in the new input deck remains unchanged. This allows the new input deck to be used to analyze the same core loading at a later burnup and presents the user with up to date burnup information when designing new core loading patterns.

![Figure 4-1: Post TRIGSIMS flowchart showing generation of burned input deck for existing loading pattern.](image-url)
While processing the output deck from TRIGSIMS, the post_trigsims code has the option to read the power distribution data generated by TRIGSIMS, which is listed as power per axial section of each fuel rod, and convert it to the Normalized Power (NP) used in determining Tech Spec compliance. This NP data is exported to a user specified text file and will be discussed in greater detail in Chapter 5. For cases where multiple control rod configurations were analyzed using the TRIGSIMS branch card, data for all branches will be exported.

**Section 4.1.2: Converting to Fixed Source**

Applying weight window-based variance reduction methods to an MCNP model of the PSBR requires that the model utilized a fixed source. The iterative source definition used in criticality mode problems is incompatible with the aggressive particle splitting techniques. By default, TRIGSIMS creates fixed source definition for each MCNP input deck by first running the ADMARCH code to find power distribution then writing a source definition card within MCNP. This feature was intended to decrease the number of cycles in criticality calculations which must be discarded before the source definition has fully converged. For large reactor geometries, using such a best estimate to initialize the criticality source saves considerable time and allows fewer kcode cycles to be skipped. For small reactors, such as the TRIGA, criticality mode problems converge very rapidly even with a poor initial source guess. For this reason, the source definition produced by TRIGSIMs is not useful for criticality mode problems.

TRIGSIMS’s fixed source definition is intended only as a best guess, and was never intended to be used independently of the MCNP’s iterative criticality source, making it unsuitable as a source definition to be directly applied to a fixed source simulation with weight windows. It does however provide a very easy starting point for creating a new source definition. To allow the PSBR MCNP model to be utilized with weight window-based variance
reduction, two script subroutines were prepared. The first modifies an MCNP deck prepared by TRIGSIMS to include fission rate tallies for each section of fuel in the model. The second takes the results of running this model and replaces the TRIGSIMS source definition with the more accurate MCNP tally results.

Section 4.1.3: Geometry Addition

The ability to model additional geometry beyond that included in the TRIGSIMS package was one of the most important features in this upgrade the PSBR modeling capabilities. The impact of the previous D$_2$O tank on power distribution was identified as a case that should be analyzed during the development of TRIGSIMS and has been included in the change package for all core loading since Core 53.

The earliest efforts to include the new D$_2$O tank in analysis of new core loadings focused on adding only the D$_2$O tank to the model and expanding the outer boundary of the problem accordingly. Figure 2-4 is an example of this early geometry, and such a model was used to predict the effects of the new tank on Core Loading 57. If the final purpose of the model was to be restricted only to core analysis as was the goal of TRIGSIMs modeling the previous D$_2$O tank, this limited geometry would have been a reasonable stopping point. However, since the ability to easily analyze experiments after changes in core loadings was also desired, the geometry needed to be expanded beyond the range that contributes to in core power distribution.

As an intermediate step in the implementation of geometry additions to the TRIGSIMS model, separate scripts were written to create input decks for various experimental facilities, specifically the new D$_2$O tank, the FNI, and an open pool model which included the D$_2$O tank. Maintaining scripts to adjust geometry on three different models proved to be excessively difficult during the time that the engineering design for the new D$_2$O tank was being finalized.
To simplify this effort, the D$_2$O tank geometry addition script was expanded to include the geometry of the full pool with all experimental facilities. This single model was then divided into the final sub-models discussed in Chapter 2.

**Section 4.1.4: Processing TRIGSIMS tallies outside of TRIGSIMS**

The addition of geometry discussed in the previous section occurs after TRIGSIMS has completed its execution. After adding geometry to the TRIGSIMS generated model, it is still necessary for the updated input deck to be run in MCNP. The tally definitions and output formats remain unchanged, meaning TRIGSIMS is capable of reading the results, but no means exists to force TRIGSIMS to begin execution at this point. Two methods were considered for the post-processing of TRIGSIMS tally results after geometry addition without making changes to the TRIGSIMS code itself.

The first method considered was to perform the addition of geometry in line with TRIGSIMS execution of the MCNP model. This could be achieved by changing the name of the executable called for in the TRIGSIMS input deck to a script which would perform the geometry addition before running MCNP. Other than a few extra seconds of runtime while geometry is being added, the TRIGSIMS code would function exactly the same as usual (baring any unexpected error checking). This method would allow the post_trigsims script to be used to process the TRIGSIMS output deck to determine NP in exactly the same way previously employed. The disadvantage to this method is that TRIGSIMS would continue to perform its predictor/corrector methodology to burn the core loading even though the corrector step is not utilized when analyzing a core loading at a single burnup time. Figure 4-2 shows how the standard TRIGSIMS burnup process would be modified to include geometry in this way.
To avoid the need to essentially “trick” TRIGSIMS into executing additional code, and to avoid an unnecessary burnup calculation, a script was written to process TRIGSIMS generated tallies outside of TRIGSIMS. This script essentially repeats function of both TRIGSIMS and post_trigsims, producing a summary of Normalized Powers. Processing the tally results in this way has another advantage in that additional experimental changes can be simulated by modifying the input deck, so long as the TRIGSIMS generated tallies remain unchanged. This introduces considerably more flexibility and has already proven useful when analyzing potential experiments in the core region.
Section 4.1.5: Setting up and Running TRIGSIMS and MCNP cases

Each new core loading requires numerous cases to be analyzed using TRIGSIMs and MCNP. In addition to testing limiting cases for excess reactivity and control rod worth, power distribution must be considered for a wide range of control rod positions. RSEC procedure SOP-3 outlines suggested cases to be simulated [12]. Additional cases have been added based on experience with the TRIGSIMS code and the changes in experimental facilities at the PSBR. With the addition of the New Core Moderator Assembly (and removal of the old D₂O tank), a total of 21 cases are now included. These cases are summarized in Table 4-1. Cases simulating the previous D₂O tank, denoted as XD and D-Branch cases, are no longer included. The cases for the new D₂O tank are denoted as XU and U-Branches. In this way, any confusion between which of the different tank versions is being simulated is eliminated when reviewing historical records.

<table>
<thead>
<tr>
<th>CASE ID</th>
<th>BURNUP</th>
<th>REACTOR POSITION</th>
<th>FUEL TEMP</th>
<th>CONTROL RODS</th>
<th>COMMENTS</th>
</tr>
</thead>
<tbody>
<tr>
<td>CLa</td>
<td>Last Measurement of Current Core Loading</td>
<td>R1</td>
<td>cold</td>
<td>ARO</td>
<td>Uses known core loading</td>
</tr>
<tr>
<td>Xa</td>
<td>Current</td>
<td>R1</td>
<td>cold</td>
<td>ARO</td>
<td>Excess Reactivity</td>
</tr>
<tr>
<td>a-BRANCH</td>
<td>Current</td>
<td>R1</td>
<td>cold</td>
<td>20cm (TR in)</td>
<td>TR Worth</td>
</tr>
<tr>
<td>a-BRANCH</td>
<td>Current</td>
<td>R1</td>
<td>cold</td>
<td>20cm (TR out)</td>
<td>TR Worth</td>
</tr>
<tr>
<td>a-BRANCH</td>
<td>Current</td>
<td>R1</td>
<td>cold</td>
<td>20cm (SA in)</td>
<td>SA Worth</td>
</tr>
<tr>
<td>a-BRANCH</td>
<td>Current</td>
<td>R1</td>
<td>cold</td>
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<td>SA Worth</td>
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<tr>
<td>a-BRANCH</td>
<td>Current</td>
<td>R1</td>
<td>cold</td>
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<tr>
<td>Xb</td>
<td>Current</td>
<td>R1</td>
<td>hot</td>
<td>ARO</td>
<td></td>
</tr>
<tr>
<td>b-BRANCH</td>
<td>Current</td>
<td>R1</td>
<td>hot</td>
<td>25cm (k~1)</td>
<td></td>
</tr>
<tr>
<td>b-BRANCH</td>
<td>Current</td>
<td>R1</td>
<td>hot</td>
<td>SUL</td>
<td></td>
</tr>
<tr>
<td>Xu</td>
<td>Current</td>
<td>New D2O</td>
<td>cold</td>
<td>ARO</td>
<td></td>
</tr>
<tr>
<td>u-BRANCH</td>
<td>Current</td>
<td>New D2O</td>
<td>cold</td>
<td>20cm (TR in)</td>
<td>TR Worth</td>
</tr>
<tr>
<td>u-BRANCH</td>
<td>Current</td>
<td>New D2O</td>
<td>cold</td>
<td>20cm (TR out)</td>
<td>TR Worth</td>
</tr>
<tr>
<td>u-BRANCH</td>
<td>Current</td>
<td>New D2O</td>
<td>cold</td>
<td>20cm (SA in)</td>
<td>SA Worth</td>
</tr>
<tr>
<td>u-BRANCH</td>
<td>Current</td>
<td>New D2O</td>
<td>cold</td>
<td>20cm (SA out)</td>
<td>SA Worth</td>
</tr>
<tr>
<td>u-BRANCH</td>
<td>Current</td>
<td>New D2O</td>
<td>cold</td>
<td>0cm (SA out)</td>
<td>Used only for SDM</td>
</tr>
</tbody>
</table>
Most of the cases are variations upon temperature and rod position for the reactor in the open pool at the current burnup state. A single script is used to take a base TRIGSIMs input deck that describes the loading pattern and create an input deck for each case at the present burnup. An additional input deck is created with parameters appropriate to burn the core loading as specified in SOP-3 for analysis at a later burnup. All of these input decks are generated before TRIGSIMs is executed. This allows the code used for setting up cases to be used independently of TRIGSIMs if only a single case is to be analyzed.

Following setup of the current burnup inputs, each case is run using TRIGSIMs, and the outputs saved. The results of the burnup calculation are utilized as inputs to the post_trigsims routine to create an additional TRIGSIMs input which similarly serves as the base input for burned cases. The cases for the new D$_2$O tank differ from corresponding open pool cases only by the addition of geometry. As such, the MCNP decks created by TRIGSIMs for these open pool cases are used as the base input for the geometry addition scripts.

Section 4.1.6: Loading Analysis Report Generation

Having already prepared scripts to automate the setup and execution of all analysis cases for new core loadings, it is still necessary to process the outputs into a concise report that describes all limiting cases. Even with the use of tools such as post_trigsims, this has remained a labor-intensive process of copying data from text files into spreadsheets, sorting by normalized power, and identifying any cases where the safety system IFE is not in the position of MEPD. A new report generation method was written to process the data after TRIGSIMs execution. This script reads power distribution data from the tally files generated by MCNP in the same way as TRIGSIMS and extracts criticality data from the output deck. By bypassing the TRIGSIMS
output decks in this way, the same report generation code can be used for both open pool cases, which run in TRIGSIMs, and D₂O tank cases which do not.

Each case is processed individually to determine MEPD, NP, and the maximum permitted value of the Limiting Safety System Setting (LSSS), the setpoint at which the reactor will SCRAM due to high fuel temperature. The results are appended to a report file in a comma separated value (CSV) format. This allows for easy importing into spreadsheet software for formatting, data retention, and inclusion in RSEC documentation such as AP-12s for the new core loading. An example of a loading report for a modified version of Core 58A is shown in Table C-2.

Section 4.1.7: Graphical Display of Core Loading Pattern

A core loading display and status board application developed at the Oregon State TRIGA Reactor (OSTR) was provided to PSBR staff. This application contained a graphical display of the OSTR loading pattern, an editable list of fuel element positions, and an interface for recording notes about various experimental facilities [29]. An example of the default view of this application is shown in Figure 4-3. As can be seen, the geometry is very different from that of the PSBR. All of the source code for this application was written in the same version of the Python scripting language as used elsewhere in this thesis, making it an easy starting point for modifications to suit the PSBR.
By utilizing elements of both the OSTR code and geometry provided by the PSBR staff, a new display application was developed. While the initial attempt to adapt the code to the PSBR focused only on making changes to the original, this new effort started from scratch and re-used only code that was evaluated to be necessary. This allowed a full graphical display of the PSBR core loading to be very quickly coded, without the coding incompatibilities encountered by the PSBR staff. Several new elements were added to describe the PSBR such as 8.5 and 12wt% fuel, Instrumented Fuel Elements, FFCRs, Transient Rod, Dry Tubes, and Grid Plate Supports. The original CSV input deck format of the OSTR code was preserved. To get data for PSBR core loadings into the graphical display, a new script was written to convert the data from a TRIGSIMS input deck into the CSV input format. Flexibility was left in this code to allow additional data, such as normalized power or elemental power density, to be added and displayed.
at a later time. By starting from the TRIGSIMS input deck format, this code will be compatible with any previous loading input since 2006, as well as any future core loading pattern.

*Figure 4-4: PSBR Status Board Display showing Core Loading 58A. Fuel elements are identified by PSBR element serial number.*

An example of the core display for Core 58A is shown in Figure 4-4. This diagram is very similar to the core loading maps previously used at the PSBR such as Figure 2-2 which are created by manually updating the entries for each fuel element in a graphics file. For future core loadings, this automatically generated map is expected to replace the manually formatted one.
Section 4.1.8: Rapid Plotting of TRIGSIMS generated MCNP Input Geometry

An ongoing challenge when working with the MCNP model of the PSBR is the considerable time required for MCNP’s built in geometry plotter to display geometry. Without modifications, the TRIGSIMs generated model takes roughly two minutes to display. When adding geometry to the model for experimental facilities, a highly iterative process, this can quickly add up to hours of time spent waiting. A similarly long delay is also present at the beginning of each MCNP run as cross section libraries are read for each material. While these cross sections are needed for transport calculations, they are unnecessary for debugging geometry.

To speed up geometry setup and debugging, a script was prepared which creates a copy of an input deck to be plotted with the material definitions for all fuel segments removed. In this way, the time to plot the geometry of the core and experimental facilities can be reduced to just 10 seconds. This quick plotting routine creates an input deck that cannot be used for transport calculations (accidentally or otherwise) but allows the geometry to be plotted almost instantaneously. Upon closing the geometry plotter, the temporary input deck is automatically deleted to prevent its unintended use. The original input deck remains unmodified.

In addition to speeding up the speed with which geometry is plotted, the quick plot subroutine also adds a greater degree of consistency to the plotting. A com file is generated by the subroutine containing a set of definitions defining the color and other default settings to be used when plotting different materials. Despite being a very minor change, this consistent color usage makes the plotted input decks far more useful for inclusion in publications. Previously any addition of geometry could result in changes to the colors used when plotting (such as Figure 2-3 and Figure 2-4). An example of this consistent color scheme can be seen in Figure 2-6.
Section 4.2: Automatic Generation of New Core Loading Patterns Through Random Shuffling

A large portion of this work has focused upon automation and removal of the need for human interaction with the process of core analysis. The scripts that resulted from this development provided numerous means of interacting with TRIGSIMS and the MCNP inputs it generates that were not previously available. To expand this work beyond making simple modifications to inputs and making outputs easier to read, a method was coded to serve as a tool in designing new loading patterns in addition to analyzing them.

The diffusion code ADMARCH included in TRIGSIMS is often overlooked when running core analysis cases, since executing MCNP takes up the majority of the runtime and is the source of all power distribution and criticality results output by TRIGSIMS, however it is possible to run TRIGSIMS using only ADMARCH to find a rough estimate of $k_{eff}$ and power distribution. [30] This estimate can serve as a confirmation of engineering judgment when shuffling fuel before committing to the longer runtime of MCNP. Because the diffusion code requires a discretization and homogenization of the geometry, the results are not as accurate as the Monte Carlo based simulation, but they may still be used to estimate the effect of changes to known core loadings, especially as intermediate steps between running MCNP for a full core loading analysis.

The ability to estimate reactivity effects quickly, with TRIGSIMS execution in ADMARCH only mode taking roughly 90 seconds, presents an opportunity to utilize the existing fuel management tools ways not otherwise practical. Runtime of the ADMARCH only mode can be further decreased by giving TRIGSIMS a fake directory of MCNP cross sections. Since MCNP is not being used in this mode, there is no need to commit runtime to preparing to
generate an MCNP input. This is able to reduce runtime in ADMARCH mode down to roughly one second. Using ADMARCH, the effect of loading changes on excess reactivity for dozens or even hundreds of scenarios can be analyzed in less than the time needed to test a single case in MCNP. The most time-consuming part of this process became the time spent by the user generating loading pattern changes and modifying input decks. If the user interaction with an iterative design process for core loading changes could be minimized, the process could be further accelerated.

The limitations placed upon core loading configurations of the PSBR by Technical Specification and facility procedures are generally not difficult to comply with when making small to moderate changes to the loading pattern. The normalized power and MEPD restrictions are almost impossible to reach when the size of the core is roughly 100 fuel elements. These limitations were more problematic in the past when the installation of multiple new 12wt% elements allowed the core to operate with roughly 70 fuel elements. The prohibition upon placing a 12wt% element next to an in-core water hole can be achieved by simple visual inspection of the core map. With these parameters being far from limiting, the Excess reactivity limitation can easily be used as a singular design parameter. To this end, an algorithm was developed and scripted to make small changes to the core loading model and evaluate them solely on the effect of change in excess reactivity.

The loading change generation algorithm functions by first reading a TRIGSIMS input deck and generating a list of fuel elements internal to the code. A second list of fuel elements is generated by excluding elements from the full list if they meet certain criteria that would make them less than ideal for shuffling, creating a list of available elements. These criteria and their basis will be discussed in detail later. Next, an element is selected at random from the list of
available elements. To mitigate the generating non-symmetric core loadings which may exhibit flux tilting, a line of symmetry is enforced on an axis from the Transient Rod to the Safety Rod. The code next verifies that the randomly selected element has a corresponding symmetric element available for shuffling. If it does not, a new element is selected at random and the process is repeated. Once an eligible pair of elements is found, a second pair is identified using the same procedure, and the positions of the two sets of elements are switched. A new TRIGSIMS deck is generated with the updated pattern is generated and run in the ADMARCH only mode.

A single iteration of the random shuffle algorithm can be run in under 2 seconds. To turn this random shuffle method into a useful tool, several user tunable parameters are defined. The two most important are the total excess reactivity increase goal and the minimum increase per shuffle. When being used to design a new core loading, multiple shuffle iterations are likely to be required. An outer loop of the code begins by running the original input and finding keff before generating a random change. If the changed core increases reactivity by at least the user specified minimum and less than a user specified maximum, the changes are kept. If the reactivity change is outside of the specified range, the code goes back to the previous pattern. This process is repeated until the total excess reactivity increase goal is achieved.
Several restrictions are placed upon which elements are eligible to be shuffled. Satisfying one or more of the following criteria excludes a fuel element from the list of available elements sampled by the code. Control rods are the first of the non-eligible types. For the 4 control rods, there are a total of 8 positions in the PSBR’s lower grid plate where a control rod could be placed. Any such changes are significantly beyond the scope of a normal core loading.
change and would require a new design for the control rod drive mechanics. As such, this is not viewed as a shuffle that should be considered for an automated process.

Dry tubes are the second non-eligible rod definition within TRIGSIMS. The two dry tubes currently in the core are limited in where they can be placed in the core because of the need for them to extend to the surface and remain accessible. Beginning with Core 58A, a second set of dry tubes were added to the core model in TRIGSIMS to simulate the front grid plate supports. These dry tubes are modeled as having negligible internal diameter and are in a fixed position within the core structure.

The final two categories of rods that cannot be shuffled are a user specified list of elements, and elements with burnup less than a user specified limit. Preventing movement of elements based on burnup is necessary to avoid the random placement of low burnup 12wt% elements next to waterholes, a requirement of Technical Specifications. Without this exclusion, there is a high probability than any newly added fuel would be moved to the B-Ring of the core. Providing the ability to manually exclude specific elements was originally intended to prevent instrumented fuel elements from being shuffled, especially as their burnup increase, to keep them in or near the position of MEPD, but can also be used to ensure that elements manually added to the core for a specific purpose remain where intended.

As an example of the effectiveness of this methodology, a test case was run to increase the reactivity of Core 58A by $0.50 through random shuffling. A minimum reactivity increase of $0.05 and a maximum of $0.10 per shuffle was specified. In addition to limiting the effect of any single change, the maximum increase value also sets a limit on the overshoot of the total reactivity increase target. Table 4-2 summarizes the fuel moves identified, as well as their corresponding reactivity changes. A total of 9 successful iterations were required to develop this
new core loading pattern, adding $0.53$ of excess reactivity to the core. An MCNP simulation of the final loading pattern confirmed the reactivity increase from the original case to be $+0.55\pm0.04$.

Table 4-2: Summary of shuffled elements and reactivity changes

<table>
<thead>
<tr>
<th>ITERATION NUMBER</th>
<th>SOURCE ELEMENT 1</th>
<th>SOURCE ELEMENT 2</th>
<th>TARGET ELEMENT 1</th>
<th>TARGET ELEMENT 2</th>
<th>REACTIVITY INCREASE</th>
<th>TOTAL REACTIVITY INCREASE</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>126</td>
<td>123</td>
<td>66</td>
<td>l-15</td>
<td>$0.05$</td>
<td>$0.05$</td>
</tr>
<tr>
<td>2</td>
<td>138</td>
<td>137</td>
<td>229</td>
<td>228</td>
<td>$0.09$</td>
<td>$0.13$</td>
</tr>
<tr>
<td>3</td>
<td>138</td>
<td>137</td>
<td>241</td>
<td>240</td>
<td>$0.05$</td>
<td>$0.18$</td>
</tr>
<tr>
<td>4</td>
<td>24</td>
<td>36</td>
<td>209</td>
<td>207</td>
<td>$0.07$</td>
<td>$0.25$</td>
</tr>
<tr>
<td>5</td>
<td>138</td>
<td>137</td>
<td>215</td>
<td>219</td>
<td>$0.06$</td>
<td>$0.31$</td>
</tr>
<tr>
<td>6</td>
<td>131</td>
<td>122</td>
<td>204</td>
<td>203</td>
<td>$0.05$</td>
<td>$0.36$</td>
</tr>
<tr>
<td>7</td>
<td>236</td>
<td>235</td>
<td>229</td>
<td>228</td>
<td>$0.06$</td>
<td>$0.42$</td>
</tr>
<tr>
<td>8</td>
<td>24</td>
<td>36</td>
<td>206</td>
<td>212</td>
<td>$0.06$</td>
<td>$0.48$</td>
</tr>
<tr>
<td>9</td>
<td>203</td>
<td>204</td>
<td>119</td>
<td>69</td>
<td>$0.05$</td>
<td>$0.53$</td>
</tr>
</tbody>
</table>

REFERENCE: MCNP confirmation of total increase: $0.55\pm0.04$

Figure 4-6 shows a map of the newly developed core loading. A total of 559 iterations were required to generate this loading, with the entire simulation taking 30 minutes. This time is comparable to the amount of time that would be needed to manually configure the input deck for the final configuration generated. An advantage to producing potential loading patterns with this random shuffle method is that estimating the reactivity is a built-in part of the process. To manually setup and run only the 9 successful iterations would be far more time consuming than the time needed to automatically run all 559 cases run here.
Figure 4-6: Core Loading 58A (top) and an example core map generated by applying the random shuffle algorithm (bottom) to Core 58A. Fuel elements are identified by PSBR element serial number.

The net reactivity increase vs. successful iteration is shown in Figure 4-7. By restricting both the minimum and maximum allowed reactivity increases, each iteration produces relatively consistent progress towards the final core loading. On the lower reactivity end, this serves to
minimize the total number of elements that must be moved to achieve the core loading. Limiting the maximum reactivity insertion per shuffle prevents a single pair of low burnup elements from being moved into a position where they could potentially generate enough power to force a reduction in the LSSS.

To further demonstrate the need to place both upper and lower limits upon the reactivity increases accepted, Figure 4-8 shows the net reactivity increase of all cases. The net reactivity of the current accepted case is also shown in red. The 17th iteration in this example was found to increase reactivity by $0.37. This was achieved by moving elements 238 and 239, two of the lowest burnup 12 wt% elements in the core, into the B-Ring. The burnup of these elements is high enough that the elements are considered to be eligible to be shuffled, however placing them
in the B-Ring would almost certainly make one of them the MEPD element and force the LSSS to be reduced. This figure also shows that the majority of shuffles result in a reactivity decrease. This makes sense because the core loading is already designed in order to achieve a high excess reactivity with the limited quantity of fuel elements already installed. The B-Ring currently consists of the six newest 8.5 wt% elements in the core and the C and D-rings contain 12 wt% fuel.

Figure 4-8: Net reactivity changes of all iterations for random shuffle example
The distribution of reactivity changes, relative to the most recent accepted iteration, is shown as a histogram in Figure 4-9. Each bin has a width of $0.05$. All cases greater than +$0.05$ and less than -$0.30$ are grouped into overflow and underflow bins, respectively. The most common range for reactivity changes produced by random shuffle was between -$0.05$ and $0.00$. The next largest group was reactivity increases of up to $0.05$. These two bins show that the majority of the cases tested showed only a small reactivity change ($\pm 0.05$).

![Figure 4-9: Distribution of reactivity changes produced by random shuffle algorithm](image)

The results shown in this example are consistent with extensive testing performed during the development of the core shuffle code. The ability to utilize fuel elements located in storage racks was also implemented and tested. This capability showed only limited improvements to the capability to add reactivity by random shuffling, since much of the fuel in the storage racks was removed from the core after Core Loading 53H due to high burnup. It is also undesirable to
use burned fuel from the storage racks because much of it has been inspected in preparation for shipping to return it to the Department of Energy (DOE), and it’s use in future core loadings may preclude its shipment. Nonetheless, the option to include storage rack fuel in random shuffles will be preserved in the event that burnup of the in-core fuel increases to a level that would justify its use before a shipment can occur.

A major limitation of the random shuffle method as it has been implemented is that it relies strictly upon the total reactivity feedback. Considerations such as changes in power peaking are not evaluated for intermediate steps. As a result, the final core loading produced by this method could be a core loading that more detailed MCNP simulations show does not comply with Technical Specifications requirements. This method is also limited in that it only considers the open pool position. Experimental facilities such as the D₂O tank and FNI are not considered.
Chapter 5. Development and Analysis of a New Core Loading Pattern for Operations at the New Core Moderator Assembly

The core loading of the PSBR is routinely changed to account for fuel burnup. Refueling is often scheduled such that new fuel is added during the biennial fuel inspection, thus limiting the number of maintenance outages. Each new core loading is analyzed to ensure compliance with technical specifications requirements. Additionally, the impacts of experimental facilities with significant reactivity contributions are analyzed. Using the current core loading and an early version of the new model, the effects of the new moderator assembly on the core is analyzed. Based on these results, design objectives for a new core loading are prioritized.

Section 5.1: Analysis of New Moderator Assembly with Core 57

Using the updated model of the new core moderator assembly, the current core loading, Core 57, was analyzed. This was done both to be able to compare results to the work of Uçar, and to begin to estimate the changes to the core loading that will be necessary for routine use of the new design. Modifications and comparisons were made to an input deck for Core Loading 57 which contains burnups as of January 29, 2016. This input was used in the design of Core 57, and full results for power distribution and reactivity are available for comparison. The parameters investigated in this analysis are: power distribution, D₂O tank worth, excess reactivity, and shutdown margin. Power distribution will be quantified as both Normalized Power (NP), “the ratio of the power of a fuel element to the average power per fuel element” and Maximum Elemental Power Density (MEPD), “the power density of the element in the core producing more power than any other element in that loading” [2]. It should be noted that all work shown within this section was performed early in the process of reconstructing the model
of the moderator assembly, as shown in Figure 2-4. As such, all results are preliminary, and may not agree exactly with results of the more detailed finalized model.

**Section 5.1.1: Power Distribution**

Coupling the reactor to the D$_2$O tank was found to shift power towards the front and sides of the core. Elements near the front of the core saw up to a 43% increase in power production. Elements near the back of the core, far from the tank, decreased power by as much as 12%.

Element I-17, the current safety system instrumented fuel element remained MEPD element with a power of 17.05kW (~4% decrease from open pool position). Figure 5-1 shows a map of Core Loading 57 and includes the ratio of power production per element between the model including the new D$_2$O tank and the open pool model.

![Figure 5-1: Map of Core Loading 57 showing ratio of power production per fuel element (NP) at New D$_2$O Tank vs. Open Pool showing a shift towards the front and outside of the core.](image-url)
Section 5.1.2: D₂O Tank Worth

Using a modified (graphite reflectors removed) version of Core Loading 53, Uçar predicted the new core moderator assembly would have a reactivity worth of $1.13. [20] [5] Using the same method of calculating the excess reactivity contribution, the worth of the moderator assembly for Core 57 can be found as:

\[ \text{Worth} \ (\$) = \frac{k_{\text{eff}}(D_2O) - k_{\text{eff}}(\text{Open Pool})}{k_{\text{eff}}(D_2O) \cdot k_{\text{eff}}(\text{Open Pool}) \cdot \beta_{\text{eff}}} \]

where \( \beta_{\text{eff}} \), the effective delayed neutron fraction for the PSBR, is assumed to be equal to 0.007 [1]. Using the simulation results presented in Table 5-1, the worth of the new D₂O tank was found to be $1.47. This is significantly higher than the value predicted previously. A likely cause of this change which will be investigated is the difference in core loading. Core 54 introduced 6 fresh 8.5 wt% elements to the core and replaced many of the periphery fuel elements with less heavily burned fuel from the fuel storage racks. Core 56 added a fresh instrumented 12 wt% element, I-17. Analysis of Core 53G with the new D₂O tank model is expected to produce results consistent with previous analysis.

Table 5-1: Model k\text{eff} results for all rods fully withdrawn comparing Core 57 in the Open Pool and D₂O Tank Positions

<table>
<thead>
<tr>
<th>MODEL</th>
<th>( k_{\text{eff}} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>CORE</td>
<td>1.05192 ± 0.0002</td>
</tr>
<tr>
<td>CORE AND D₂O TANK</td>
<td>1.06345 ± 0.0002</td>
</tr>
</tbody>
</table>

Section 5.1.3: Excess Reactivity

The total excess reactivity of the core and moderator assembly can be calculated as:

\[ \text{Excess} \ (\$) = \frac{k_{\text{eff}} - 1}{(k_{\text{eff}} \cdot 1)\beta} = \frac{1.06345 - 1}{1.06345 \cdot 0.007} = 8.52 \]
This estimate significantly exceeds the Technical Specifications (Tech Spec or TS) limit of $7.00 for excess reactivity. Based on over a decade of use, the MCNP inputs generated using the TRIGSIMS code are known to ever estimate the excess reactivity of the core and the worth of the control rods. To get a more accurate estimate of the expected excess reactivity, we can add the predicted worth of the D$_2$O tank, $1.47$, to the measured excess reactivity of Core 57 in the open pool position ($5.51$) [31]. This gives us an expected excess reactivity of:

$$\text{Expected Excess Reactivity} = \text{measured} + \text{modeled increase} = 6.98$$

which is just (nominally) under the Tech Spec limit.

Section 5.1.4: Shutdown Margin

Technical Specifications for the PSBR require that the reactor can be made subcritical by at least $0.25$ with the most reactive control rod (generally the safety rod) stuck in its most reactive position (fully withdrawn). Core Loading 57 has a measured shutdown margin of $3.13$. To determine the expected shutdown margin of Core 57 at the new D$_2$O tank, several simulations were completed. First, the open pool and new D$_2$O tank models were run with the safety rod fully withdrawn and all other rods SCRAMed. The results of these simulations are shown in Table 5-2. The shutdown margin can be found from this result as:

$$SDM = \frac{1 - k_{eff(SUL)}}{(1 * k_{eff(SUL)}) * \beta}$$

<table>
<thead>
<tr>
<th>MODEL</th>
<th>$K_{EFF}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>CORE</td>
<td>0.98150 ± 0.0002</td>
</tr>
<tr>
<td>CORE AND D$_2$O TANK</td>
<td>0.99767 ± 0.0002</td>
</tr>
</tbody>
</table>
Table 5-3 summarizes the predicted shutdown margins for Core 57, as well as the measured value. The predicted value for the core in the open pool, $2.64, underestimated the measured value by 15.4%. Applying the ratio of the measured to predicted values to the predicted SDM at the new D_2O tank, we find a biased estimate of the SDM of $0.39. Since the model predicted a conservative estimate of the shutdown margin for Core 57 in the open pool position, it is expected that any error in the D_2O tank position will also be conservative. Since both the biased and unbiased predictions for this scenario are greater than the minimum value required by Technical Specifications, the core is expected to satisfy the SDM requirement.

<table>
<thead>
<tr>
<th>MODEL</th>
<th>SDM</th>
</tr>
</thead>
<tbody>
<tr>
<td>CORE 57 (MCNP)</td>
<td>$2.64</td>
</tr>
<tr>
<td>CORE 57 AND MODERATOR (MCNP)</td>
<td>$0.33</td>
</tr>
<tr>
<td>CORE 57 (MEASURED)</td>
<td>$3.12</td>
</tr>
<tr>
<td>CORE 57 AND MODERATOR (BIASED)</td>
<td>$0.39</td>
</tr>
</tbody>
</table>

**Section 5.2: Design of Core 58 for First Criticality at the New D_2O Tank**

The analysis presented in the previous section showed that Core 57 can be used with the new core moderator assembly without violating Technical Specifications. However, based on previous experience using the TRIGSIMS code, uncertainty on reactivity predictions for control rod worth are greater than the margins between predicted values and Tech Spec limits for both excess reactivity limit and shutdown margin. There is a high probability that experimental measurements made using Core 57 and the new moderator assembly would show that these
Technical Specifications requirements are not satisfied, thus requiring the core loading to be changed before the new core moderator assembly can be utilized. To avoid any delays when restarting the reactor after the installation of the new moderator assembly, two new core loadings were designed, Core 58 and Core 58A. The fuel arrangement of these two cores are identical. The differences between the loadings is mostly administrative. Core 58 was analyzed using the old D$_2$O tank. Core 58A was modeled using the new D$_2$O tank and included two aluminum supports for the upper gridplate within the core region.

**Section 5.2.1: Core 58**

Core Loading 58 was installed in March 2018 at the conclusion of a biennial fuel inspection. Such fuel inspections and loading changes are normally performed during the summer but was conducted early to avoid scheduling conflicts with installation of the new core moderator assembly. The change was presented in AP-12 2018-2 [11]. Unlike most core loading changes, one of the goals of Core 58 was to decrease excess reactivity of the core. Elements 238 and 239, the lowest burnup 12wt% elements in the core, were moved from the core face to the E-Ring at the back of the core. This was done to shift power production away from the Safety Rod before transitioning to Core 58A, thus limiting the effect upon shutdown margin. A map of Core 58 is shown in Figure 5-2.

As part of this core loading change, several other fuel moves of lesser reactivity effect were performed to prepare for the new D$_2$O tank. The in-core dry tubes were relocated from the core periphery to the E-Ring. This move was performed to make space for the upper gridplate supports which, in the new tower, are located in the positions previously occupied by the dry tubes. An older instrumented fuel element, I-15, was moved to the C-12 position. When coupling to the new D$_2$O tank, this position will be surrounded by the tank on two sides and is
expected to see the greatest fractional change in power production in the core as a result. The normalized power in this location is negligible, but this spare IFE was viewed as potentially useful in the event that fuel temperature measurements adjacent to the tank were desired. A second 12wt% element, 221, was intended to be placed in the symmetric position on the other end of the core face, however this element was removed from service as a result of the fuel inspection performed before establishing the core loading. An 8.5wt% element, 66, was used in its place, with TRIGSIMS analysis showing no significant changes to the core loading as a result [32].

![Figure 5-2: Core Loading 58 Map](image-url)
Section 5.2.2: Core 58A

The change analysis for Core 58A introduced the structural supports for the upper gridplate and included the new D$_2$O tank [33]. The predicted reactivity effect of this change with the reactor in the open pool was found to be minimal. Adding the structural supports did however have an effect upon power distribution. While the safety system element I-17 remained in the MEPD position for all analyzed cases in Core 58, the addition of aluminum supports shifted flux inwards at the front side of the core enough so that the analysis cases with the safety rod at the upper limit caused Element 135 to become the MEPD element for both the open pool and D$_2$O tank cases. Under these circumstances, it was necessary to reduce the LSSS to no higher than 627°C. (The setpoint was already set to 575°C, so no adjustment was required.)
During installation of the new D$_2$O tank, measurements were performed using dummy fuel elements to verify the space between fuel and the tank, as shown in Figure 5-4. This dry fit measurement was repeated several times as adjustments to the beamtubes and tank were made. The final test fit also included loading weights to the tower equivalent to the mass of fuel in a standard core loading to ensure that the tower is properly settled.
The clearance between the tank and fuel was found to be greater than the 0.25 inches specified in the design on all five sides, as shown in Figure 5-5. The geometry of the new tank was updated within the MCNP model, and the analysis was repeated using the most up to date burnup information. The updated clearances and burnup did not have a significant effect upon power distribution or reactivities, but the estimated shutdown margin increased by 18%. This makes the shutdown margin less limiting upon core loading design and can be viewed as a favorable change.
Measurements of reactivity for Core 58A in the open pool were consistent with those for Core 58, confirming that no unexpected change occurred as a result of replacing structural elements such as the tower and gridplates. The reactivity worth of the new D$_2$O tank was measured to be lower than expected. Before accounting for the as built geometry, the model predicted that the tank would have a reactivity effect of +$1.06$. The as built model decreased this to +$0.97$. Experience with the TRIGSIMS model of the old D$_2$O tank has shown that the measured worth of the tank is consistently 120% of predicted, and it was assumed that the new tank could exceed the model prediction by a similar amount. The measured reactivity worth of the new core moderator assembly was found to be just +$0.85$. Because this is only the first core loading used with the new tank, it is impossible to be sure that the model will consistently overestimate the worth of the tank, but this measurement will be used to provide a biased estimate of tank worth for future core loadings as a point of reference.
### Table 5-4: Predicted and measured reactivity values for New D₂O Tank

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<thead>
<tr>
<th>CASE</th>
<th>REACTIVITY</th>
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</thead>
<tbody>
<tr>
<td>ORIGINAL PREDICTION</td>
<td>$1.06</td>
</tr>
<tr>
<td>“AS-BUILT” PREDICTION</td>
<td>$0.97</td>
</tr>
<tr>
<td>MEASURED REACTIVITY</td>
<td>$0.85</td>
</tr>
</tbody>
</table>

#### Section 5.3: Development of Core Loading 59 Using New Software Tools

The schedule for fabrication and installation of the new core moderator assembly placed significant constraints upon the design of Core Loadings 58 and 58A. The software to reliably include the new moderator assembly was developed concurrently with the final design of the D₂O tank, and many intermediate steps in the analysis were still performed manually. Both of these core loadings were designed at the same time using the most recent burnup data available. The time between the design of the cores and installation of Core 58 was sufficiently short that this did not present a problem. A greater issue was the extensive operations history and burnup of Core 58 in order to complete all planned operations before shutting down for installation of the new D₂O tank.

To minimize any impact of burnup upon the predictions for core loadings, the time between design/analysis and installation of the core loading should be as short as possible. If a full core loading change were suggested here to replace the currently installed Core 58A, the newly designed loading is unlikely to be installed until at least 6 months have passed. Instead of presenting a fully developed and analyzed core loading, a clearly outlined plan for the design process of Core 59 will be presented.

The primary goal of Core Loading 59 will be to increase excess reactivity as near as possible to the $7.00 limit. This will counter fuel burnup and allow the reactor to continue to
operate for several more years before additional loading changes. Upon initial startup, the excess reactivity of Core 58A was $5.08 (including $0.85 from the new D₂O tank) [34]. This has been adequate to allow for high power operations even at the FNI, but additional burnup will soon force power reductions, especially when coupled with xenon poisoning. The Shutdown Margin specification is expected to be the most limiting restriction upon the core loading when coupled to the New D₂O tank. Because this will be only the second core loading designed for the new core moderator assembly, there is inadequate data to show if the model consistently over or under predicts reactivity parameters. This design will assume that the ratio of predicted to measured values for Core 58A will be valid for future core loadings while at the same time attempting to leave sufficient margin in case they are not.

The first step in the design of Core 59 will be to use TRIGSIMs to deplete the fuel definitions in the Core 58A input to match the core at the time the design simulations are started. Ideally, this should be done at a time when control rod calibrations and excess reactivity calculations have been performed recently, thus eliminating the need to rely upon simulated estimates of reactivity change due to fuel burnup. After depleting fuel materials using TRIGSIMS, the post_trigsims code will be used to create an input deck with current burnup. The input decks for Core 58A at both the present time and at the time of the most recent control rod calibration/excess reactivity evaluation should both be run. The difference in reactivity between these provides the estimated effect of burnup since the most recent evaluation of excess reactivity. A target for the total amount of reactivity to add can be found by subtracting the last measured excess and estimated burnup from the $7.00 limit.

With an estimate of the amount of reactivity addition calculated, the next step is likely to be manually addition of unburned fuel. This will provide a greater reactivity addition than any
single random shuffle could hope to achieve, as well as providing greater control over the effects upon power distribution. The availability of fresh fuel at the PSBR is limited, so its addition to the core is only made when necessary. Recent testing of the random shuffle algorithm has shown that the amount of reactivity addition needed to reach the excess limit is greater than can be achieved by random shuffling alone, even if the burned fuel inventory not currently in the core is considered.

Excluding instrumented fuel elements, no unburned 8.5wt% elements are available for installation at present. The currently installed IFE, I-17, remains the element of MEPD in most control rod configurations [33], meaning there is no need to install a new IFE at this time. This leaves installing new 12wt% fuel as the most viable option. SOP-3 specifies that fresh 12wt% fuel should be added to the C-Ring of the reactor core [12], and Technical Specifications preclude its installation in the B-Ring where it would have a greater reactivity effect or adjacent to an in-core water hole. The recommended core location for the installation of new 12wt% elements is in the C-Ring in the positions on either side of the Transient Rod, with the elements in those positions (228 and 229) being swapped with 8.5wt% elements that would then be removed from the core. The current worth of the Transient Rod is $3.02 [34], well below the $3.50 limit. Placing fresh fuel in these positions will also serve to boost flux at the IFE without negatively impacting the shutdown margin.

The reactivity change between the core with added 12wt% elements and the current burnup model should be calculated, and this difference can be subtracted from the total reactivity addition target to find a target for reactivity addition from random shuffling. This input deck can now be used as the base input for the random shuffle algorithm, with the calculated reactivity addition from random shuffling as one of the execution parameters. The random shuffle script
will produce a final TRIGSIMS input deck that can then be used to perform a full core analysis using TRIGSIMS and MCNP through the automated core analysis script to generate a loading analysis report. This process for the design of Core 59 is presented graphically in Figure 5-6.

Figure 5-6: Core 59 design flowchart
Chapter 6. Summary, Conclusions, and Recommendations

Section 6.1: Summary

New software has been prepared in order to allow the fuel management tools in use at the PSBR to model the newly installed core moderator assembly. Other experimental facilities such as the FNI and FFT were also included to ensure that these models are readily available for analysis of experiments after the installation of new core loadings. Emphasis has been placed on maintaining the methods of analysis historically used for new core loadings while streamlining the process to remove much of the need for manual configuration and processing of input and output decks.

Following extensive testing, a set of variance reduction parameters have been generated for all experimental facilities to allow simulations of experiments to be performed up to 128 times faster than the analog simulation. For the FNI and FFT, default settings within the ADVANTG code were found to be acceptable for the generation of these weight window libraries. For the new beamports, the presence of streaming paths along multiple angles prevented any of the default quadrature schemes tested from working for all beamports. To avoid this limitation, a custom quadrature set was created for each beamport that includes a vector roughly parallel to the beamport to account for streaming. This was found to allow the ADVANTG code to generate weight window libraries that proved effective in accelerating MCNP simulations and avoiding “long history” problems.

To complement the streamlining and automation of the core loading analysis process, a new random shuffle algorithm was implemented. This method utilizes the ADMARCH code, already present within the TRIGSIMS package, to rapidly determine the effect of random changes to the core loading. The shuffle algorithm selects an element randomly from the core.
If the element meets user specified eligibility requirements, it is swapped with another randomly selected element. The reactivity effect of the change is estimated by running the TRIGSIMS code in ADMARCH mode. This method has proven extremely effective in identifying core loading changes that result in small increases in excess reactivity. This new method is expected to be a useful means of designing new PSBR core loadings to better utilize existing fuel inventories.

After showing that the new core moderator assembly may cause Core Loading 57, the loading installed at the time, to exceed the excess reactivity limitations within PSBR Technical specifications, the updated model was used to design two new core loadings: 58 and 58A. Core 58 was the last loading before the new core moderator assembly. This loading slightly decreased the excess reactivity of the core and shifted power production away from the Safety Rod. Core 58A utilized the same loading pattern as Core 58 but included the new upper grid plate supports of the new reactor tower as well as analysis for the new D$_2$O tank. Core 58A is currently installed and has enough excess reactivity to continue to operate for another year before refueling. A plan has been presented within this thesis for the design and analysis of Core Loading 59 using the newly developed tools. Core 59 will be the first PSBR core loading created using the new random shuffle algorithm.

**Section 6.2: Conclusions**

Software to include the new core moderator assembly within the existing fuel management framework has been developed. The newly developed software automatically adds the geometry of all experimental facilities at the PSBR to the models produced by TRIGSIMS. Previously, it was necessary to manually add the geometry for each experimental facility any time that an experiment was to be simulated. This led to experiments being simulated with outdated burnup
and loading pattern definitions simply to avoid large setup times. By automating the inclusion of this geometry, any experimental facility can be easily including in simulations for the analysis of new core loadings or experimental design.

The preparation and execution of input decks as well as processing of code outputs for core analysis has been automated. In the past, each of the input decks for TRIGSIMS was manually configured and executed. This often resulted in the analysis of core loadings taking several weeks longer than necessary as the next input deck to describe the core loading cases was only configured after the previous case had been completed. Scripting of this process is a simple but very substantial change to new core analysis at the PSBR.

Variance reduction parameters for the new beamports have been generated which accelerate simulations by up to a factor of 70. Development of these weight window libraries required extensive testing of both built in features of the ADVANTG code as well as development of a custom quadrature set for each beamport. These custom quadratures add a low-weight vector roughly parallel to the beamport to an establish quadrature set. This enabled the DENOVO code to find an adjoint flux for each beamport adequately accounted for streaming down the beamtube. Of note is the fact that a highly accurate solution for the adjoint flux is not needed in order to generate weight windows. While correctly solving for the adjoint is preferable, an inaccurate solution is adequate provided it identifies the streaming path. The weight windows generated were extensively tested to show not only their speedup capabilities, but to ensure that they did not bias the results of tallies within MCNP.

Variance reduction parameters for the FNI and FFT have been generated which accelerate simulations by greater than a factor of 100. Unlike the beamports, these experimental facilities to not involve long streaming paths and could be easily described using default quadrature sets.
This considerable increase in speed of simulations, combined with the increased ease of modeling these facilities with new core loadings, will make it far easier for future researchers to perform detailed simulations when designing experiments to be irradiated at the PSBR.

A method of automatically shuffling the core to increase excess reactivity has been developed and implemented. This shuffle algorithm was carefully written to give the user control over which fuel elements can be shuffled and which are not automatically moved in order to avoid creating configurations that could challenge Technical Specifications limits. Testing of this algorithm has demonstrated its ability to identify a series of small reactivity increases that would be prohibitively time consuming to manually design and analyze. The end result is the creation of new core loading patterns that utilize existing fuel inventory to increase the excess reactivity of the core. In the future, this is expected to prove invaluable as only a limited inventory of unburned fuel is available. The automatic generation of new core loading patterns could help to keep the PSBR operating for some time until new fuel can be acquired.

Multiple core loading patterns have been designed and analyzed to accommodate the new core moderator assembly. Core Loading 57 was shown using simulations to be nominally compliant with limitations upon excess reactivity and shutdown margin. Core 57 was never tested with the new D₂O tank. Instead, a set of new core loadings, Core 58 and Core 58A were designed and analyzed. These core loadings decreased excess reactivity to allow for considerable errors in predictions for the new moderator assembly without approaching technical specifications limits. Reactivity measurements for Core 58A show that model predictions for the new D₂O tank overpredict the tanks effect, thus Core 57 would almost certainly have complied with the excess limitation, however by designing the more conservative Core 58A, it was
ensured that the PSBR was able to be restarted after the installation of the new moderator assembly without delays for core redesigns to reduce excess reactivity.

A plan to develop a new core loading for long term use at the new core moderator assembly has been presented. Core loading 59 will add a large amount of excess reactivity to return the core to near the Technical Specifications limit. This will allow the PSBR to operate for four to six years without adding additional fuel. This core design will also serve as a final test of all of the software written to support the new core moderator assembly, incorporating all aspects of the codes prepared as part of this work.

Through this work, the fuel management software used at the PSBR has been updated to allow for modeling of new experimental facilities and to streamline the process of designing and analyzing new core loading patterns. Using a new method to adapt quadrature sets to the recently installed beamports, adjoint driven weight window variance reduction techniques have been applied to the PSBR model. Extensive testing has demonstrated that this method allows Monte-Carlo simulations of the PSBR to be performed up to 96 times faster without biasing the results. The software and variance reduction parameter libraries produced are expected to remain in use at the PSBR for years to come.

**Section 6.3: Recommendations for Future Work**

In the process of updating the fuel management and neutronics model of the PSBR to accommodate the new core moderator assembly, multiple ideas for investigation of the model or potential improvements were discovered which could be useful if investigated further.

A comparison of the forward flux predictions from DENOVO with measured values should be made. The custom quadrature set employed to generate weight windows can be expected to
introduce an error into the values calculated for the forward and adjoint fluxes along the axis of the beamtube. When using adjoint based weight windows, such an error is compensated for by the Monte-Carlo calculation. It is possible that this error is small enough that the deterministic predictions are accurate enough to be useful on their own. If this is the case, the ADVANTG code could be used as a front end for DENOVO estimate flux and dose rate in the Beamlab and elsewhere.

If a reliable MCNP model of the cold neutron guides can be developed, then a set of weight windows should be developed for the guides. The custom quadrature sets developed for the other beamports will not work for a curved guide. The suggested solution is to utilize the finest mesh size and highest quadrature order possible within the ADVANTG code. This will likely require the use of computer resources beyond those currently available at the RSEC, as even the rather simple custom quadratures presented here utilized upwards of 64GB of RAM.

It should be verified that predicted vs. measured values for reactivity worth of the D$_2$O tank are consistent across core loadings. This will require at least one more core loading change, which is not practical at this time based on the excess reactivity of Core 58A. D$_2$O tank worth was consistently underestimated for the old tank. The only available data for the new tank indicates that the actual worth of the tank is 87% of the model prediction. If the model consistently overpredicts the reactivity effect of the new tank, then it can be more easily used to design core loadings that maximize excess reactivity as near as possible to the Technical Specifications limit.

Additional validation of the power distributions predicted by the TRIGSIMS model for the core at the new D$_2$O tank should be performed as the reactor schedule permits. Methods presented in the SAR can be employed to experimentally verify NP predictions at the new D$_2$O
tank by relocating I-14 within the core. Change in flow introduced by the new tank may impact the accuracy of the steady state versions of this test, but the pulse method should not be significantly impacted. Such measurements made in the past have shown good agreement with TRIGSIMS predictions.

The features introduced here to streamline the analysis process based on user experience should be carried forwards into a replacement for the TRIGSIMS code. TRIGSIMS relies upon outdated software that is no longer supported and in some cases is incompatible with current versions (SCALE 6.1 and 6.2 are incompatible with TRIGSIMS). Ideally, any replacement would maintain the ability to generate input decks for MCNP but would not rely upon codes requiring licensing to perform necessary functions.
Works Cited


[34] PSBR Staff, CCP-11 Core Evaluation: 8/14/2018, University Park, 2018.
Appendix A: Example ADVANTG Input Deck

- method: cadis
- outputs: mcnp silo response
- denovo_verbose: True
- denovo_quadrature: userdefined
- denovo_quad_file: WW6_split_NBP1.txt
- mcnp_input: 58A_FixedSource2NBP1.in
- mcnp_tallies: 32
- mcnp_sb_type: none
- anisn_library: 27n19g
- mesh_x: -301 301
- mesh_x_ints: 201
- mesh_y: -101 401
- mesh_y_ints: 175
- mesh_z: -171 -101 101
- mesh_z_ints: 20 50
- mcnp_mxspln: 100
- mcnp_wwCollapse_factor: 10
Appendix B: Additional Figures
Appendix B.1: Adjoint Flux Maps

Figure B-1: Adjoint Flux Map for NBP1
Figure B-2: Adjoint Flux Map for Cold Neutron Beam Port
Figure B-3: Adjoint Flux Map for NBP2
Figure B-4: Adjoint Flux Map for NBP3
Figure B-5: Adjoint Flux Map for NBP4
Figure B-6: Adjoint Flux Map for the Reactor at the FNI (with MCNP geometry plot for context)
Figure B-7: Adjoint Flux Map for the Reactor at the FFT (with MCNP geometry plot for context)
Appendix B.2: Forward Flux and Response Density Maps

Figure B-8: Forward Flux Map for New Beamports
Figure B-9: Response Density Map for NBP1
Figure B-10: Response Density Map for Cold Neutron Beam Port
Figure B-11: Response Density Map for NBP2
Figure B-12: Response Density Map for NBP3
Figure B-13: Response Density Map for NBP4
Appendix B.3: Weight Window Map Sets

Figure B-14: Base MCNP Geometry for Tank Sub-model
Figure B-15: Weight Window Map for NBP1 Generated by ADVANTG
Figure B-16: Weight Window Map for Cold Neutron Beamport Generated by ADVANTG
Figure B-17: Weight Window Map for NBP2 Generated by ADVANTG
Figure B-18: Weight Window Map for NBP3 Generated by ADVANTG
Figure B-19: Weight Window Map for NBP4 Generated by ADVANTG
Appendix B.4: Collapsed Weight Window Map Sets

Figure B-20: Collapsed Weight Window Map for NBP1 Generated by ADVANTG
Figure B-21: Collapsed Weight Window Map for Cold Neutron Beamport Generated by ADVANTG
Figure B-22: Collapsed Weight Window Map for NBP2 Generated by ADVANTG
Figure B-23: Collapsed Weight Window Map for NBP3 Generated by ADVANTG
Figure B-24: Collapsed Weight Window Map for NBP4 Generated by ADVANTG
## Appendix C: Additional Tables

### Table C-1: Detailed Summary of Runtime Data for Weight Window Testing

<table>
<thead>
<tr>
<th>Beamport</th>
<th>Run</th>
<th>1MW Flux [n/(cm²*s)]</th>
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<th>Runtime [s]</th>
<th>FOM</th>
<th>Speedup</th>
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<td>NBPC (includes Cold Source Moderator, built in XS)</td>
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Table C-2: Example Loading Report after Importing to Excel

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<td>650.0</td>
</tr>
<tr>
<td>T58Au1</td>
<td>I-17</td>
<td>1.653</td>
<td>15.747</td>
<td>I-17</td>
<td>1.653</td>
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<tr>
<td>T58Au2</td>
<td>I-17</td>
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<td>16.782</td>
<td>I-17</td>
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</tr>
<tr>
<td>T58Au3</td>
<td>I-17</td>
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<td>17.327</td>
<td>I-17</td>
<td>1.819</td>
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<tr>
<td>T58Au4</td>
<td>I-17</td>
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<td>16.162</td>
<td>I-17</td>
<td>1.697</td>
<td>650.0</td>
</tr>
</tbody>
</table>

Reactivity Summary:
- Temperature defect: -3.8271
- D2O Tank Worth: 0.8450
- TR Worth R1: 3.3488
- TR Worth D2O: 3.1485
- SA Worth R1: 3.7429
- SA Worth D2O: 4.0325
### Table C-3: Full Cosine and Weight Data for NBP1 Quadrature Set Octant

<table>
<thead>
<tr>
<th>Ordinate Number</th>
<th>X Cosine</th>
<th>Y Cosine</th>
<th>Z Cosine</th>
<th>Original Weight</th>
<th>Modified Weight</th>
</tr>
</thead>
<tbody>
<tr>
<td>Original set</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>1</td>
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<td>0.16047800000000</td>
<td>0.97390637282646</td>
<td>0.06667140000000</td>
<td>0.0666629841418</td>
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<tr>
<td>2</td>
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<td>0.19197800000000</td>
<td>0.86506326065786</td>
<td>0.07472570000000</td>
<td>0.074715988871</td>
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<tr>
<td>3</td>
<td>0.19197800000000</td>
<td>0.46347600000000</td>
<td>0.86506326065786</td>
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<td>0.074715988871</td>
</tr>
<tr>
<td>4</td>
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<tr>
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</tr>
<tr>
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<tr>
<td>10</td>
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<tr>
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<td>0.05909673136173</td>
</tr>
<tr>
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</tr>
<tr>
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</tr>
<tr>
<td>14</td>
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<td>0.88107700000000</td>
<td>0.14887671849554</td>
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</tr>
<tr>
<td>15</td>
<td>0.15469100000000</td>
<td>0.97668200000000</td>
<td>0.14887231238548</td>
<td>0.05910480000000</td>
<td>0.05909673136173</td>
</tr>
<tr>
<td>NBP1</td>
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<td>0.63661676071142</td>
<td>0.000001745327408</td>
<td>0.00000000000000</td>
<td>0.00013651409477</td>
</tr>
</tbody>
</table>

### Table C-4: Cosines for Adding New Beamports to Quadrature Set Octant

<table>
<thead>
<tr>
<th>Beamport</th>
<th>X Cosine</th>
<th>Y Cosine</th>
<th>Z Cosine</th>
</tr>
</thead>
<tbody>
<tr>
<td>NBP1</td>
<td>0.77118032889635</td>
<td>0.63661676071142</td>
<td>0.000001745327408</td>
</tr>
<tr>
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<td>0.000001745328681</td>
</tr>
<tr>
<td>NBP2</td>
<td>0.00001745329252</td>
<td>0.99999999969538</td>
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</tr>
<tr>
<td>NBP3</td>
<td>0.41405786877686</td>
<td>0.91025055946116</td>
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</tr>
<tr>
<td>NBP4</td>
<td>0.78187967150587</td>
<td>0.62342936968133</td>
<td>0.000001745329317</td>
</tr>
</tbody>
</table>
Vita

Andrew Bascom was born in New Jersey, The United States of America on January 12, 1990. He received his Bachelor of Science degree in nuclear engineering from The Pennsylvania State University in May 2012. He continued his graduate studies at The Penn State University while working as a Senior Reactor Operator at the University’s research reactor. Andrew received his Master of Science in December 2015. He completed his PhD titled “Improvements to the PSBR Fuel Management and Neutronics Model in Support of the New Core Moderator Assembly” in August 2019.