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Department of Mechanical and Nuclear Engineering

**MODELING AND DESIGN OF A NEW CORE-MODERATOR ASSEMBLY AND
NEUTRON BEAM PORTS FOR THE PENN STATE BREAZEALE NUCLEAR
REACTOR (PSBR)**

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by
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ABSTRACT

This study is for modeling and designing a new reactor core-moderator assembly and new neutron beam ports that aimed to expand utilization of a new beam hall of the Penn State Breazeale Reactor (PSBR). The PSBR is a part of the Radiation Science and Engineering Facility (RSEC) and is a TRIGA MARK III type research reactor with a movable core placed in a large pool and is capable to produce 1MW output. This reactor is a pool-type reactor with pulsing capability up to 2000 MW for 10-20 msec. There are seven beam ports currently installed to the reactor. The PSBR's existing core design limits the experimental capability of the facility, as only two of the seven available neutron beam ports are usable.

Modeling of the new PSBR design was accomplished by completing neutronic and thermal-hydraulics analyses by using MCNP5, ANSYS Fluent, Gambit, TRIGSIMS, MURE, and Burned Coupled MCNP simulation tool. The finalized design features an optimized result in light of the data obtained from neutronic and thermal-hydraulics analyses as well as geometrical constraints. A new core-moderator assembly was introduced to overcome the limitations of the existing PSBR design, specifically maximizing number of available neutron beam ports and mitigating the hydrogen gamma contamination of the neutron beam channeled in the beam ports. A crescent-shaped moderator is favored in the new PSBR design since it enables simultaneous use of five new neutron beam ports in the facility. Furthermore, the crescent shape sanctions a coupling of the core and moderator, which reduces the hydrogen gamma contamination significantly in the new beam ports. A coupled MURE and MCNP5 code optimization analysis was performed to calculate the optimum design parameters for the new PSBR. The optimum dimensions of the crescent-shaped moderator tank were calculated as 76.2 cm in radius and 48.2 cm in height. All new beam ports were directed to the core center and the optimum distance between the reactor core face and each new beam port was calculated. At optimum distance,

which varied between 10 cm and 18 cm for individual beam ports, maximum thermal neutron beam with suppressed fast neutron and gamma components were realized.

Thermal-hydraulics analysis of the new design was achieved using ANSYS Fluent CFD code. In the current form, the PSBR is cooled by natural convection of the pool water. The driving force for the natural circulation of the fluid is the heat generation within the fuel rods. The convective heat data was generated at the reactor's different operating powers by using TRIGSIMS, the fuel management code of the PSBR core. In the CFD modeling, the amount of heat generated by the fuel is assumed to be transferred totally into the coolant. Therefore, the surface heat flux is applied to the fuel cladding outer surface by considering the depleted fuel composition of each individual fuel rod under a reference core loading condition defined as; 53H at 1MW full power. In order to model the entire PSBR reactor, fine mesh discretization was achieved with 22 millions structured and unstructured computational meshes. The conductive heat transfer inside the fuel rods was ignored in order to decrease the computational mesh requirement. Since the PSBR core operates in the subcooled nucleate boiling region, the CFD simulation of new PSBR design was completed utilizing an Eulerian-Eulerian multiphase flow formulation and RPI wall boiling model. The simulation results showed that the new moderator tank geometry results in secondary flow entering into the core due to decrease in the cross-flow area. Notably, the radial flow improves the local heat transfer conditions by providing radial-mixing in the core. Bubble nucleation occurs on the heated fuel rods but bubbles are collapsing in the subcooled fluid. Furthermore, the bulk fluid properties are not affected by the bubble formation. Yet, subcooled boiling enhances the heat transfer on the fuel rods. The CFD predicted bulk fluid temperature in the hot channel of new reactor core was 11 °C higher than the measured bulk fluid temperature in the hot channel of existing reactor core. The maximum bulk fluid temperature in the hot channel is estimated as 88 °C by the CFD model, which is less than the

saturation temperature of water coolant at reactor's operating conditions. In addition, the CFD predicted maximum clad outer surface temperature and calculated maximum fuel temperature in the new PSBR core are estimated as 175 °C and 482 °C, respectively. These temperatures are well below the safety limits of the PSBR.

Five neutron beam ports are designed for the new reactor. Four new experimental techniques and facilities are added to the existing Neutron Imaging (NI) and Neutron Transmission facilities in the new beam hall. Triple-Axis Spectrometer (TAS), Conventional and Time-of-Flight (TOF) Neutron Depth Profiling (NDP), Neutron Powder Diffraction (NPD), and Prompt Gamma Activation Analysis (PGAA) are the proposed new techniques. The geometrical configuration, filter and collimator system designs of each neutron beam ports are selected based on the requirements of the experimental facilities. A cold neutron beam port which utilizes cold neutrons from three curved guide tubes is considered. Therefore, there will be seven neutron beams available in the new facility. The neutronic analyses of the new beam port designs were achieved by using MCNP5 code and Burned Coupled Simulation Tool for the PSBR. The MCNP simulation results showed that thermal neutron flux was increased by a factor of minimum 1.23 times and maximum 2.68 times in the new beam port compared to the existing BP4 design. Besides total gamma dose was decreased by a factor of 100 in the new neutron beam facility. Therefore, this study assures both the thermal-hydraulics safety and neutronics performance improvements in the new PSBR design.

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Chapter 1

INTRODUCTION

Neutrons have been used as a valuable research tool in several areas of science, including engineering, physics, materials science, biotechnology, etc. since the neutrons discovery by Chadwick in 1932 [1]. Neutrons very rarely remain in a free state. They are mostly bound to a nucleus. Hence, neutron generation is only possible when this bound is broken. In general, free neutrons are obtained from any of these three different ways: radioactive sources, spallation sources, and research reactors. An example for the first group of sources is some certain isotopes that can undergo spontaneous fission and emit neutrons. Cf-252 is the most commonly known spontaneous fission source. Its half-life is 2.6 years and the neutron emission rate is $\sim 2.3 \times 10^6$ neutrons/second/microgram. The latter category is the spallation neutron source (SNS). The SNS is an accelerator-based neutron source in which protons are accelerated to high energies to hit a target material. Neutrons are emitted from the target material after the interaction. The third group is research reactors in which neutrons are produced due to fission reaction by U-235. After the discovery of nuclear fission in 1938, it was realized that more neutrons can be generated from chain reaction and this led to the development of nuclear reactors. Nuclear reactors are mainly built for electricity generation, and hence are called power reactors. However, some power reactors are utilized for research and development. These reactors are known as research reactors. Research reactors operate at lower temperature and they need less fuel compared to power reactors. Their primary purpose is to supply neutrons for analysis and testing of materials, radioisotope production, neutron scattering, and research and education. Research reactors are utilized in many fields including fusion research as well as in nuclear industry, advanced materials development, environmental sciences, technology development and in nuclear

medicine. There are about 230 research reactors currently operating in 56 countries [2]. Although most (160) of them are utilized for research and radioisotope production, they are also employed for other purposes: 60 critical assemblies, 23 test reactors, 37 training facilities, 2 prototypes and one producing electricity. Many have been operating more than 30 years. Therefore, they approach to the end of their design lifetime.

In a research reactor, neutrons are transported from the reactor core to experimental facilities via neutron beam ports (or tubes). The characteristics of the neutron beam in a beam port depend on many factors including the size and shape of the beam ports as well as the location and orientation of the beam ports with respect to the reactor core, core-moderator configuration, reflectors, and filter and collimator design in the beam ports. Each of these factors needs to be well defined in order to utilize high density of neutrons with appropriate energy and time structure to the experimental facilities.

The Radiation Science and Engineering Center (RSEC) at The Pennsylvania State University hosts many nuclear research facilities including the Penn State Breazeale Reactor (PSBR), Gamma irradiation facilities, hot cells, the radio-nuclear application laboratory, the neutron beam laboratory, the radiochemistry laboratories, the nuclear security education laboratory and radiation detection and measurement resources. Most of these research facilities are located in the PSBR building. The RSEC provides these facilities for nuclear research and education not only to faculty, staff, and students at Penn State University, but also to users from other universities, government institutions, and the industry. The Penn State Breazeale Reactor (PSBR) is the longest operating research reactor in the United States.

Motivation

The PSBR is a TRIGA Mark-III type research reactor with a movable core in a large pool, capable of generating up to 1MWt power. It is a pool-type reactor with pulsing capability up to 2000 MW for 10-20 msec. There are seven beam ports in the reactor facility, and neutron beam becomes available in only two of these beam ports when the reactor core is coupled with a drum-shaped D₂O moderator tank. However, the PSBR's current core design limits the use of all the available neutron beam ports. There are currently only two beam ports, named as #4 and #7, functional in the facility. Beam port #4 (BP4) which is located along the core centerline is primarily used for neutron radiography, radioscopy, neutron depth profiling (NDP), neutron irradiation of samples (soft error rate measurement), and slow-chopper neutron spectrometer. Beam port #7 (BP7) is utilized for neutron transmission measurements [3]. There are various experimental facilities of which some are currently in use and some are under development. The list of available and being developed experimental facilities is given in Table 1-1.

There are two major limitations in the existing PSBR design. The first one is the core-moderator assembly, as it does not allow simultaneous use of more than two beam ports in the facility. The second one is the amount of floor space in the existing beam hall. The number of experimental facility or instruments that can be installed to the beam hall cannot be increased. All the existing beam port facilities are currently installed in BP4 except the neutron transmission facility. Therefore, almost all the experiments in which thermal neutrons are utilized are conducted in BP4. However, the neutron beam in BP4 has a very high gamma component which significantly affects almost all the measurements including NDP, Time-of-flight (TOF)-NDP, Neutron Imaging, etc. [4][5].

Table 1-1: The list of existing and proposed experimental facilities in the RSEC

Current facilities:	Additional Proposed Facilities:
Neutron Irradiation	Triple-Axis Spectrometer
Neutron Radiography	Prompt Gamma Activation Analysis (PGAA)
Neutron Activation Analysis (NAA)	Neutron Depth Profiling (NDP)
Radioactive Isotope Production	Time-of-flight Neutron Depth Profiling (TOF-NDP)
Neutron Transmission Measurement	Neutron Powder Diffraction (NPD)
Nuclear Detector Testing and Calibration	
Radiation Effects Testing on Materials and Electronics	
Gamma Irradiation	

Minor improvements have been implemented to the existing facilities over the years, but none of them were adequate to advance the experimental capability of the facility. After all the limitations of the existing PSBR design are identified, the motivation of this dissertation is defined to redesign the core-moderator assembly with new neutron beam ports. This will expand the research and education capability of the facility. After these design goals are reached, the expansion of the neutron beam hall and construction of new beam laboratories will be possible. As the beam hall is expanded, significant economic and academic benefits will be attained. This will impel the Penn State University to a leading position and advance the nuclear science and engineering in U.S.

Thesis Structure

This thesis consists of seven parts, a brief description of analyzed details and relevance of each chapter are outlined as follows.

In Chapter 2, a literature review of previous neutronic and thermal-hydraulics analysis of the PSBR is given and a brief description of computational tools employed throughout the thesis is introduced.

In Chapter 3, general features of the existing design are introduced and the current limitations are explored in detail. The two major drawbacks in the current PSBR core design identified are the core-moderator assembly and the beam port configuration. The hydrogen gamma contamination in the BP4 is analyzed in this chapter. Then details of the gamma contribution including the benchmarking and MCNP modeling of the existing PSBR design are introduced and the utilization of the existing beam hall is illustrated at the end of the chapter.

The evaluation of a new PSBR core-moderator assembly design is introduced in Chapter 4. After the design requirements and design constraints are presented, a new core-moderator assembly with five new neutron beam ports is introduced and drafted in 3D using AutoCAD® software. The propensity of this proposed design is discussed at the end of this chapter.

In Chapter 5, optimum design parameters are calculated and neutronic performance of the optimum new core-moderator assembly design is analyzed using MCNP5 code.

The thermal-hydraulics analysis of the new core-moderator assembly in the optimum design is focused on in Chapter 6. The ANSYS Fluent CFD code is employed to analyze the new core-moderator assembly design in the pool. The CFD analysis results are presented with the calculated flow and temperature profiles and average void fraction distribution in the flow channels.

Chapter 7 presents the neutronic analysis of the PSBR with the new five beam port design. After each new beam port design that was utilized in the new core-moderator assembly is outlined, the neutronic performance of each beam port is calculated with MCNP simulations.

Chapter 8 concludes with a brief summary of the study further future work suggestions and comments.

Chapter 2

LITERATURE REVIEW

In this chapter, a literature review about the previous neutronics and thermal-hydraulics analyses of the PSBR is given and the computational tools are introduced. The first section is a general overview of current computational methodologies applied in neutronic analysis of nuclear reactors.

Neutronic Analysis of a Nuclear Reactor Core

Nuclear reactor cores are analyzed by using two main classes of techniques: Monte Carlo and Deterministic methods. Due to the complicated geometry of a nuclear reactor core, the only way to calculate the neutron and gamma distributions or to optimize any of the components in the reactor is to solve the reactors' linearized integro-differential Boltzmann transport (neutron balance) equation. This equation has seven dimensions: three in space, two in angle, one each for energy and time. In the deterministic method, the transport equation is discretized for space, angle, time, and energy by dividing them into groups or intervals and the resulting large algebraic equations are solved. Although these methods are very accurate for homogenous reactor cores, certain assumptions and approximations, such as cell homogenization, group cross sections, finite mesh size, etc., are required to solve the transport equation, which may oversimplify and hence introduce errors to the computation, in more complicated geometries.

In Monte Carlo methods, random history of a particle is simulated and the results of many random histories are averaged. These methods do not use any approximation to calculate any quantity, such as neutron flux or k_{eff} , in a nuclear reactor core. In general, Monte Carlo

method is applied for the solution of the neutron transport equation in a nuclear reactor. It allows the modeling of very complex geometries in three dimensions (3D) and provides very detailed results compared to the deterministic methods. The quantity of interest, such as neutron flux/current, is determined by tracking and scoring the history of each particle from its life (fixed and fission source) to its death (absorption or leakage). The physical process, absorption or scattering, at each step is estimated using probability density functions (pdf's). In order to obtain accurate results, a large number of simulations are required and the quantity of interest is obtained by averaging the result over the total number of simulations. This makes the Monte Carlo method computationally expensive. When compared; deterministic methods are usually faster than Monte Carlo methods. On the other hand, Monte Carlo methods do not have any discretization errors in space, angle, and energy. Furthermore, Monte Carlo methods do calculate the statistical error, which are not attainable in the deterministic solutions.

Although Monte Carlo methods are more accurate compared to the deterministic methods, they must employ “nonanalog” or “variance reduction” techniques to obtain practical results in certain cases. In shielding problems, variance reduction techniques are mainly employed to transport the particles to the regions of phase-space that are important to the calculation and in which relatively very few particles exist. This is a major problem in radiation shielding and it is dubbed as the “deep penetration problem”. Typical variance reduction techniques are implicit capture, geometric splitting, weight windows, and source biasing. Since these techniques bias the physical processes, such as scattering, distance-to-distance collision, particle direction after the collision, etc., unphysical results are possible. Besides that, determination of biasing parameters requires serious user experience about the physics and geometry of the problem with biasing parameters. The most recent approach to calculate the biasing parameters is to couple Monte Carlo and deterministic methods, called *hybrid* methods [6]. For source-detector shielding problems such as deep penetration problem, *adjoint flux*, or so

called “importance” function, has been used to determine the biasing parameters which allow the penetration of Monte Carlo particles toward the target region. Several automated code systems have been developed to utilize the deterministic adjoint solution to set the biasing parameters; these include MORSE/SAS4 [7], TRIPOLI [8], A³MCNP [9], etc. The most recent calculation methodologies are Cooper’s weight windows approach [10] and Forward-Weighted Consistent Adjoint Driven Importance Sampling (FW-CADIS) method [11]. These apply the forward deterministic solution to obtain space, energy and angle dependent weight windows.

In our scope, a different computational approach was followed to overcome the deep penetration problem in the PSBR beam ports. Instead of directly calculating the contribution of source neutrons and gamma rays to the target region, which was defined at the end of each beam port, the problem domain was divided into two parts and MCNP calculations were performed in two steps. In the first step, a KCODE (criticality) calculation was performed to fabricate a surface source at certain distance from the reactor source. The spatial and angular distributions of neutrons passing through the source surface were recorded in this step. The location of this surface was modeled close to the reactor core in order to reduce the relative error or variance in the tally. Then, starting from the fabricated source surface, the neutron flux tally was calculated at the detector region in the second step. This approach not only converted the criticality analysis into a fixed source problem, but also significantly reduced the computational time. The accuracy of this technique depends on the correct representation of source particles on the fixed source surface. A high accuracy here is achieved by applying *conical source direction biasing* variance reduction technique [12]. Furthermore the neutron flux tally at the target region was calculated by a deterministic approach in MCNP using *point detectors*. The advantage of this technique is that it does not bias the physics of the problem. The computational time was further reduced by

running MCNP calculations in parallel mode. Detailed analysis of this approach and the results are presented in Chapter 4.

Literature Review for the Neutronic Analysis of PSBR Core

Core and Beam Port Calculations

The PSBR core has been analyzed by different computational approaches over the years. In the past, TRICOM code system was used for the core management of the PSBR [13]. TRICOM code is based on PSU-LEOPARD, EXTERMINATOR2, and MCRAC codes [14]. TRICOM fell short to predict the core parameters with low uncertainties in the calculations because of the limitations on the codes. After that an advanced fuel management system was developed by Kriangchairporn [15] based on the HELIOS and ADMARC-H codes. Kriangchairporn improved TRICOM code system with a better geometry modeling using HELIOS and burn-up calculation using ADMARC-H. This calculation methodology was based on the deterministic methods. Recently, a code system named TRIGSIMS (TRIGA Simulator-S) was developed by Tippayakul as a fuel management tool based on Monte Carlo methods for the PSBR [16][17]. This code system has been utilized in this study in the thermal-hydraulics and neutronics simulations of the new PSBR design as a supplemental tool. It has various advanced features and many advantages compared to TRICOM code system.

TRIGSIMS provides a user interface to define the reactor geometry and material compositions. It couples the MCNP and ORIGEN-S codes. MCNP code is used as a Monte Carlo solver and ORIGEN-S is employed as a depletion solver. MCNP calculates the power distribution among the fuel elements and its output data is transferred to ORIGEN-S for the depletion calculation. A predictor-corrector scheme is applied in the depletion calculations. After the

depletion calculation is completed, material compositions are updated. Therefore, it has up-to-date burn-up history of each fuel element in the current loading pattern. Furthermore, it allows the user to place new fuel elements, to change the positions of the existing fuel elements, to change the control rod positions, and to define the moderator in the reactor geometry.

TRIGSIMS is also capable of performing depletion calculations in five axial nodes along the active length of the fuel element. The speed of the Monte Carlo code can be increased in the coupled calculation by implementing nodal-diffusion calculation based on ADMARC-H code to predict the initial source distribution. It also employs temperature and burn-up dependent cross section data sets generated for PSBR fuel rods, which increase calculations speed significantly. Moreover, parallel computation can also be implemented on MCNP calculations to further speed up the calculation process.

TRIGSIMS displays the k_{eff} and axial power distribution at five axial nodes in each fuel element from the MCNP output and burn-up data for each fuel element from the ORIGEN output. In the thermal-hydraulics analysis of the new PSBR design, heat generation rate calculated by TRIGSIMS in each fuel rod was employed.

The beam port calculations in the PSBR were first performed by Sarikaya et al. [18]. Sarikaya coupled the ADMARC-H and MCNP codes to predict the neutron and gamma flux spectra at the end of BP4. These calculations were performed in two steps. ADMARC-H code was first employed to obtain a surface source at the core-moderator interface. This source distribution was then fed into MCNP code in order to generate a boundary source at the Bi crystal exit surface for the BP4 model. Then, the neutron and gamma spectra at the end of BP4 were calculated by using this boundary source in MCNP. By implementing this approach, the neutron flux at the end of BP4 was calculated with only 14% relative error compared to the measured experimental value. In a later study, Sarikaya and Alim performed a detailed neutronic analysis in the BP4 using the same analogy with the previous study [19]. They identified the origin of the

gamma contamination problem in BP4 as the thermal neutron capture reaction by hydrogen in pool water which mainly takes place at the sides of the existing core-moderator assembly.

Optimization Studies of PSBR

The first optimization study for the existing PSBR design was performed by Wagner and Haghighat aiming to find the optimum D₂O tank size for the existing reactor core using MCNP [12]. In this study, a Watt energy spectrum was assigned to the surface in the core-moderator interface as the neutron source. A gamma source distribution to represent the core gammas was defined at the same surface. Then, a case study was performed to find the optimum moderator size. In each case, the dimensions and configurations of the tank were changed and an optimum D₂O tank size and configuration was evaluated. Although a smaller tank size was suggested as a result of this study, the optimum moderator size was not quantified.

The second optimization study was performed by Sarikaya and Alim to define the optimum location of the beam port #4 in the D₂O tank [20]. A case study was performed by calculating the total neutron flux at the beam port re-entry hall, which is the beam port entrance surface, as the BP4 location was changed in successive MCNP simulations. As the beam port was installed closer to the reactor core, an increase in the neutron beam was reported.

The expansion of the beam hall in the existing PSBR design was previously analyzed in two studies. By evaluating the limitations of the existing PSBR design, a new design was proposed by Butler [21]. In this study, two similar characteristics bearing research reactors, namely Technical University Munich (TUM) and University of Texas at Austin (UT) were examined in detail, to propose the modifications/improvements that should be made to the current core and moderator design. Butler's work also includes a source and instrument selection for the new beam hall. However, it remained conceptual because of the absence of neutronic and

thermal-hydraulics analysis to evaluate the effect of the design changes on the performance of the PSBR. On the other hand, a modular optimization code package, named MOZAIK was developed by Bekar [22]. This code is capable of determining the optimum new moderator shape for the PSBR. It evaluates the performance of the moderator by calculating the thermal neutron beam intensity at the end of BP4. TORT code was used as a transport solver in MOZAIK and two optimizers, Min-Max and Generic Algorithms (GA), were implemented in the optimizer module. MOZAIK calculated the best moderator shape for the existing core design. Then, the neutronic performance of BP4 modeled in radial direction along the core centerline in the optimized core-moderator assembly was also evaluated in this study.

Thermal-Hydraulics Analysis of the PSBR

There have been many attempts to describe the flow and temperature fields in the PSBR reactor core. Below these studies are briefly summarized in this section.

The coolant temperature profile in four flow channels within the fuel rods was first measured by Haag (1971) by using thermocouples inserted from the top grid plate. Figure 2-1 shows the locations of these flow channels in the PSBR core in 1971 [23]. The measured coolant temperature profiles were given in Figure 2-2 which was reproduced by Chang [24]. Haag observed ± 5 °C temperature variations in the readings and the measured values were averaged for two minute intervals. Noticeable drops in the coolant temperatures above the core center was clearly observed in the measured temperature profiles. As seen from Figure 2-2, the temperature drop is highest at Channel-P which is at the edge of the core and it gets smaller in the flow channels close to the center of the core. These phenomena can be explained by a strong cross-flow entering from the periphery of the core. The effect of the cross-flow is highest in the flow channels at the edge of the core and decreases in the flow channels close to the center of the core.

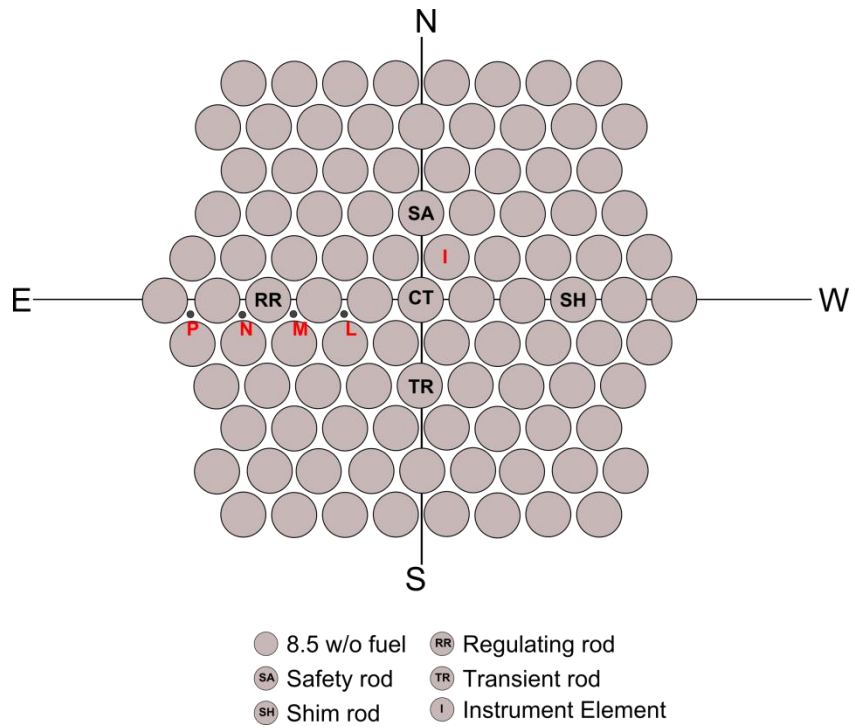


Figure 2-1: The locations of flow channels in the PSBR core in 1971

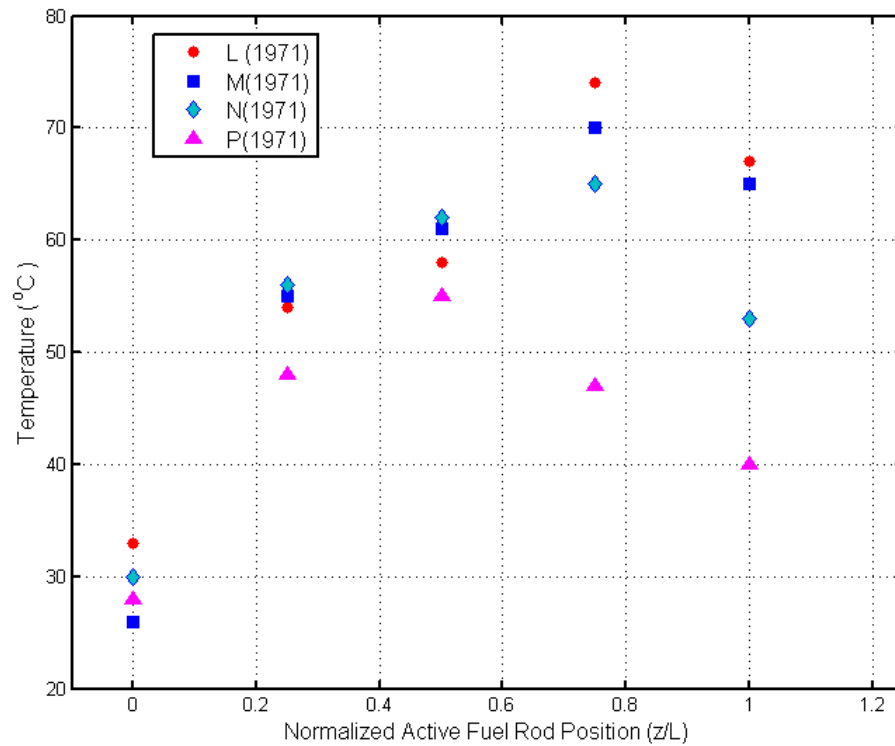


Figure 2-2: The measured temperature profiles of the bulk fluid along the active fuel rod length in four flow channels by Haag [23]

The flow profile in the PSBR core was then analyzed by Gougar (1997). He experimentally showed that 90% of the coolant is coming from the upper sides of the reactor core as a result of cross-flow [25]. In that experiment, Gougar installed shrouds around the reactor core to restrict the cross-flow. Then, the temperature change in the fuel rods both with and without shroud installation was measured. The major observation was that the fuel temperature on the upper side of the core increased more compared to the bare core with no shrouds. The fuel temperatures in the PSBR are measured by using three thermocouples installed into instrumented fuel rods and can be monitored by the reactor console. This is consistent with the observations in the measured coolant temperature profiles by Haag and it shows the effect of cross-flow on the cooling of the existing PSBR core. This was the first attempt to predict the effect of the cross-flow on the operating conditions of the PSBR. Furthermore, Gougar and then Feltus and Miller (1998) analyzed the flow profile in the PSBR core using a 3-D neutronics code, STAR (Space-Time Analysis of Reactor) and a 3-D thermal-hydraulics code, COBRA-IV (Coolant Boiling in Rod Arrays) codes [26][27]. COBRA-IV was developed for commercial power reactors cooled by forced convective flow and the natural convective flow models were not implemented to this code. That is why the analysis was not able to accurately predict the coolant temperature profile in the PSBR core which is cooled by natural convection.

In a later study, the flow field and the temperature profile around the reactor core were measured and analyzed by Chang [24] in 2004. The flow velocities in and around the core were measured by using a very sensitive sapphire jewel bearing Turbo Probe. Sapphire jewel bearing was used to improve the low velocity measurement due to its low friction. The flow measurements were made at both vertical and horizontal directions around the reactor core at 1 MW. The fuel temperatures of the instrument rods were obtained from the reactor console. Chang also created a CFD model for the core loading-51 using FLOW3D code. The total flow rate from the sides of the core (cross-flow) was predicted as $16340 \text{ cm}^3/\text{s}$ and the total flow rate from the

bottom of the core (axial flow) was predicted as $1356.4 \text{ cm}^3/\text{s}$ by the FLOW3D code. He reported that PSBR core is mainly cooled by a strong cross-flow, not axial flow. In the FLOW3D model, porous media model was applied to define the flow through the reactor core. Solid materials are treated as porous media which represents the resistance in the flow region. In this approach, heat and momentum source terms (sink parameters) need to be implemented to the Navier-Stokes equations into the code. The temperature profile in each fuel rod was determined in a separate calculation using a single fuel model. The heat generation rate in each fuel element needs to be defined in CFD model. This data was obtained from Kriangchaiporn's ADMARCH model [15]. The N-16 diffuser pump was also modeled in the CFD model. The function of N-16 pump is to dissolve the N-16 gas released from the reactor core in the pool. Four nozzles are connected to the N-16 pump and are located at 1.2 meter on the top of the core. The N-16 pump automatically runs when the reactor power becomes higher than 200 kW. Chang observed that N-16 pump mixes the flow around the reactor core and makes it unsteady. The measured flow velocity in the vertical direction is 0.5 m/s at above 1.25 m of the center of the core while the N-16 pump is off and is 0.2 m/s at the same measurement location while the N-16 pump is on. However, flow measurements inside the core were not quite successful.

The accuracy of porous medium approach directly depends on the accuracy of the source terms defined in the modified Navier-Stokes equations. It is not possible to precisely calculate these source terms in the PSBR core, especially in the top and bottom grid plates due to their complicated geometries. Therefore, porous medium approach is not precise to predict flow and temperature profile in the core flow channels. Figure 2-3 shows the comparison between the measured temperature profiles by Haag and the Flow-3D CFD predicted temperature profiles of coolant in the four flow channels (L, M, N, and P). The CFD predicted coolant temperature profiles showed great discrepancies from the measurements in channels M, N and P. These results

clearly show that porous medium approach is not suitable for the thermal-hydraulics analysis of PSBR.

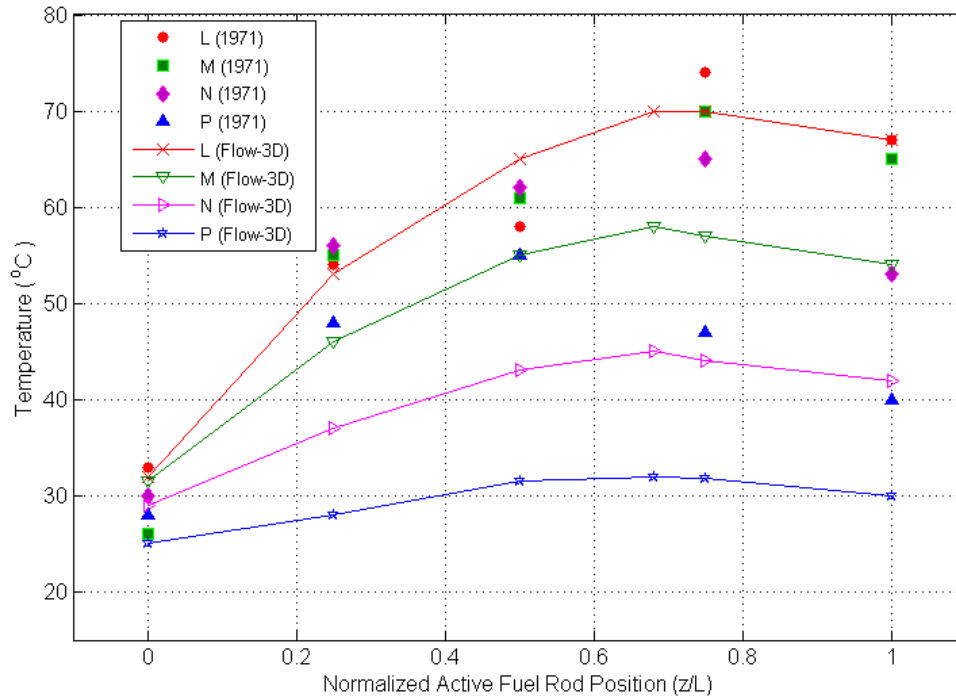


Figure 2-3: Comparison of temperature profiles in four flow channels between the measurements (1971) and Flow-3D CFD simulation

A more recent study for the thermal-hydraulics analysis of the PSBR was performed by Karriem (2011). In this study, the PSBR core and moderator design were analyzed using Cobra-TF (COolant Boiling in Rod Arrays-Two Fluid) code, named CTF, which was developed by Reactor Dynamics and Fuel Management Group (RDFMG) at the Pennsylvania State University [28]. CTF is a subchannel analysis code, which was developed mainly for thermal-hydraulics analysis of Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR). It was later employed in the analysis of PSBR. In the CTF simulations of the PSBR, the input data for the heat generation rate in each fuel rod was obtained from TRIGSIMS output. CTF code was first

tested for existing design and the results were compared with the measurements obtained at hottest channel (R16). CTF simulation results were deviated by 10% from the measurements.

In the second part of the analysis, Karriem investigated the effect of new moderator shape on the flow and temperature distribution in the new PSBR core. The flow pattern was calculated in different subchannels starting from core-moderator interface to the center. It was reported that the cross-flow on top of the new moderator through the reactor core would be similar to the cross-flow determined for the existing design. The other important outcome of the study was that the cross-flow would be higher on the edge of the core compared to the center.

Review of the Computational Tools

In this section, a brief overview was given for computational tools used in this study.

MCNP

MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. This code is developed by the Los Alamos National Laboratory. It allows users to model very complex 3-dimensional (3D) geometries using cells bound by first and second degree surfaces. The cells are defined by intersections, unions, and complements of the regions bounded by the surfaces. Time dependent problems can also be analyzed by the MCNP code.

MCNP uses continuous energy nuclear and atomic data libraries. Nuclear data libraries are produced for the modeling of neutron interactions, neutron induced photons, photon interactions, neutron activation, and thermal scattering. Neutron libraries are evaluated from Evaluated Nuclear Data File (ENDF) system [29], Advanced Computational Technology

Initiative (ACTI) [30], Evaluated Nuclear Data Library (ENDL) [31], Evaluated Photon Data Library (EPDL) [32], and Activation Library (ACTL) [33]. The evaluated data are processed into proper format for MCNP using several codes such as NJOY [34]. More than 800 neutron data tables exist for approximately 100 different isotopes and elements for different temperatures. Beside that photon interaction tables are available for all the elements. For electron and some photon interaction, photonuclear data tables are utilized.

MCNP is capable of calculating the effective multiplication factor (k_{eff}) for fissile systems. It has several built-in tally definitions for particle current, particle flux, and energy deposition. All tally results are normalized to be per starting particle except the criticality problems, which are normalized to be per fission. The units of tallies are defined by the units of the source. The variance in the tallies is calculated by extensive statistical analysis. The variance in a MCNP simulation is calculated as follows. In a Monte Carlo simulation, each particle has a score contribution to the tally. For i^{th} history, the tally contribution is denoted as x_i . If the particle or its daughter never reaches the tally region, then $x_i = 0$. The mean or expected value of x is calculated as

$$\langle x \rangle = \int_0^{\infty} xp(x)dx. \quad (2-1)$$

In this equation, $p(x)$ is probability distribution function (PDF) which defines the probability of any history that will contribute a score to the tally between x and $x+dx$. Since $p(x)$ is not known, MCNP estimates $\langle x \rangle$ by averaging the scores of N particles.

$$\bar{x} = \frac{1}{N} \sum_{i=1}^N x_i \quad (2-2)$$

As the number of histories goes to infinity ($N \rightarrow \infty$), $\bar{x}$ reaches to $\langle x \rangle$. For a large N , the standard deviation is approximated as

$$S^2 = \frac{1}{N-1} \sum_{i=1}^N (x_i - \bar{x})^2 \approx \overline{x^2} - \bar{x}^2, \quad (2-3)$$

where

$$\overline{x^2} = \frac{1}{N} \sum_{i=1}^N x_i^2. \quad (2-4)$$

The estimated variance of the average score $\bar{x}$ is

$$S_{\bar{x}}^2 = \frac{1}{N} S^2. \quad (2-5)$$

The relative error R in a MCNP simulation is then calculated as

$$R = S_{\bar{x}}/\bar{x}. \quad (2-6)$$

The relative error R is a measure of the efficiency of the MCNP simulation. It should be as small as possible, less than 1% in flux tallies and 5% for point detectors. One way of achieving this is to increase the number of histories in the MCNP simulation since R is proportional to $1/\sqrt{N}$. Additionally, MCNP prints a tally fluctuation chart (TFC) at the end of the simulation. This chart gives an indication of tally convergence and the efficiency of the calculation. 10 statistical checks are performed in this chart.

The other important quantity that is a measure of the efficiency of a MCNP simulation is the figure of merit (FOM) which is defined as

$$FOM = \frac{1}{R^2 T}. \quad (2-7)$$

In this equation, T is the simulation time and it is proportional to the number of histories N. Since $R^2 \sim 1/N$, FOM should stay relatively constant during the MCNP simulation.

ANSYS Fluent

ANSYS Fluent is a general purpose computational fluid dynamics (CFD) code for modeling fluid flow, heat transfer, and chemical reactions in complex geometries. It contains very broad physical modeling capabilities for steady-state or transient, incompressible or compressible, laminar or turbulent flows, Newtonian or non-Newtonian, and ideal or real gases as well as multiphase flow, chemical reactions, radiation and particulate dynamics. The code can solve flow problems using both structured and unstructured meshes. It supports almost all mesh types including triangular, quadrilateral, tetrahedral, hexahedral, pyramid, prism and polyhedral. ANSYS workbench also allows users to import CAD geometry and mesh it automatically using ANSYS meshing module. Dynamic refinement or coarsening of the mesh based on the flow solution is possible.

ANSYS Fluent offers extensive turbulent flow models including two equation models and Reynolds-stress models with numerous extensions such as automated wall treatment. In addition, it provides scale resolving turbulence models, such as large-eddy simulation (LES), detached-eddy simulation (DES), and scale-adaptive simulation (SAS) models. For immiscible multiphase flows, volume-of-fluid model (VOF), discrete phase model (DPM), dense discrete phase model (DDPM), and discrete element particle (DEM) models are offered by the code depending on the simulated problem. For interpenetrating fluid or phases, Eulerian multiphase flow model to capture the exchange of mass, momentum and energy are used by the code. This includes models for phase change due to cavitation, evaporation, condensation and boiling.

ANSYS Fluent has a very powerful graphical user interface (GUI) as part of the ANSYS workbench, which allows the user to easily prepare the geometry, define the models and generate graphics, animations and report for the results. More detailed information about the ANSYS Fluent code is provided by the ANSYS [35].

MURE

MURE (MCNP Utility for Reactor Evolution) is a software package written in C++ object oriented language to perform reactor time-evolution using MCNP. Due to its object-oriented nature, C++ objects give great flexibility to the users to define the geometry, materials, neutron source, tallies, etc. in the problem domain. MCNP inputs are automatically generated by MURE code depending on the user's inputs. The advantage of MURE code is the ability to make global changes to the component dimensions and locations.

MURE code is mainly developed for time evolution of reactor fuels and burn-up calculations. Using MCNP results, particle fluxes and mean-cross sections for a given reactor composition are obtained and then a numerical integration is performed to determine new material compositions for the successive MCNP calculations. MURE code has very extensive capabilities. It is very easy to define temperature-dependent cross section in the code. MURE code can also be used for coupled neutronic and thermal-hydraulics calculations for nuclear reactors. For the thermal-hydraulics analysis, Cobra-EN code is employed [36][37]. Cobra-EN is a subchannel analysis code for the steady-state and transient analysis. It provides an interface to NJOY code for easy cross section data generation. MURE also provides a GUI (graphical user interface) to visualize the depleted materials, reaction tree of each isotope, etc. It allows the user to use source distribution calculated by a previous MCNP run to reduce the calculation time.

Chapter 3

EXISTING PSBR DESIGN

In this chapter, the existing PSBR design including core, moderator, tower, and beam ports are introduced. Since the PSBR is a TRIGA-type research reactor, some technical information about the TRIGA reactor family is briefly laid out first. Then, the deficiencies of the existing PSBR design are explained. Two major problems were identified as geometrical deficiencies and gamma contamination of the neutron beam in BP#4 (BP4). After benchmarking the gamma problem, the design of beam port #4 (BP4) had recently being modified by installing a single crystal sapphire filter in order to eliminate the prompt gamma-rays from the neutron beam. In this section, the effect of the sapphire filter on the neutronic performance of BP4 was analyzed both by using gold activation analysis technique and computer simulations using MCNP5 code. Then, the energy spectra as well as the total thermal and total epithermal flux calculated from the MCNP simulations were compared with the measurement results obtained with slow chopper time-of-flight spectrometer [38]. The MCNP modeling approach and simulation results in the existing BP4 are used in the MCNP modeling of new beam port design and comparison. Therefore, this chapter forms the bases of the neutronic analysis of the new PSBR design.

TRIGA Reactor Characteristics

TRIGA is an acronym of Training, Research, Isotopes, and General Atomic. TRIGA-type reactors were first introduced in 1955 by the U.S. Company General Atomics (GA). TRIGA reactor is most widely used research reactor type in the world. There are 66 TRIGA reactors in use or under-construction in 24 countries as of today [2]. There are three TRIGA models that have been produced:

- Mark I: underground pool without beam tubes
- Mark II: above ground tank with several beam tubes
- Mark III: above ground oval tank with movable core.

All the TRIGA models are still in use all around the world. In general, TRIGA reactors:

- are inherently safe,
- are operationally flexible and relatively inexpensive,
- allow a large variety of experiments,
- operate with low enriched uranium.

The TRIGA core consists of 60-120 cylindrical fuel elements about 38 mm (1.5 inches) diameter with stainless-steel cladding. All the TRIGA models can be operated up to 2000 kW via natural convection. Additionally, Mark II and III models can be operated between 2000 kW and 3000 kW with forced convection. There is a new TRIGA type reactor [39] out of this category. This new generation TRIGA reactor located in Pitesti, Romania operates with up to 14 MW steady-state power and is cooled by forced convection.

In early core designs, fuel elements and control rods were vertically arranged on circular grid plates. However, recent TRIGA core designs employ hexagonal grid plates which allow enhanced thermal-hydraulics characteristics of the newer TRIGA reactor.

A typical fuel element in TRIGA reactors is a matrix made of a U-ZrH_x compound with strong negative temperature coefficient [39]. U-ZrH_x fuel material has several advantages compared to the aluminum cladding plate-type fuel elements used in other research and test reactors. The negative temperature coefficient of the TRIGA fuel provides inherent reactivity feedback which allows safe operation of reactor during both steady-state and transient conditions. It results from Doppler broadening of the neutron cross sections, moderator, and the ZrH_x up-scattering. Along with that, U-ZrH_x is chemically stable and can safely be quenched at about 1200 °C in water. While aluminum is the preferred cladding material in plate type-fuel elements,

stainless steel cladding (SST) is employed on TRIGA fuels since stainless steel and its alloys provide high clad integrity up to 950 °C. On the other hand, the aluminum cladding melts and fails at about 650 °C.

Most commonly used TRIGA fuel elements are produced with less than 20% enrichment. This fuel type is called standard SST fuel element. Criticality of a typical TRIGA reactor is achieved with 57 standard SST fuel elements and ~2 kg U-235. For higher than 100 kW power operation, more fuel elements and graphite reflectors are added to the core. TRIGA reactors are operated with mixed cores bearing different fuel elements because U-235 consumption is very low. To compensate the reactivity loss due to U-235 consumption, one fuel element is added to the core in every 2 years. Therefore, approximately 20 to 40 fuel elements are added in 40 years operation of a typical TRIGA reactor.

There are also 70% enriched TRIGA fuel elements which are called FLIP (Fuel Lifetime Improved Program). Due to nonproliferation concerns, FLIP fuel elements in research reactors have been gradually replaced with low enriched fuel elements. For this purpose, General Atomics certified a low enriched fuel with high uranium loading (30 wt-%) to replace the FLIP fuel. Recently, a TRIGA fuel with a smaller diameter is developed for higher than 3 MW power operations.

Penn State University Breazeale Nuclear Reactor (PSBR)

The PSBR is a TRIGA Mark-III type research reactor. It is a water cooled reactor design operating with up to 1 MWt steady-state power. The core operates at a depth of ~5.5 m (18 feet) in the reactor pool, which has dimensions of 9.14 m length, 4.27 m width and 7.21 m depth. The pool contains 270,000 liters demineralized and filtered water, providing the necessary shielding, reflection, and the cooling for the reactor. The water is stagnant and driven by natural circulation

to cool the reactor. The existing reactor core has 102 cylindrical fuel elements, 3 fuel follower control rods, and 1 air follower rod (transient rod) which are vertically oriented in a hexagonal array.

PSBR fuel elements contain homogenous U-ZrH_{1.6} mixture in which uranium is enriched up to 20%. There are two fuel element types categorized by the uranium content in the core: 8.5 wt-% and 12 wt-%. The active length of the fuel elements is 15 inches. There are three type fuel-filled (follower) control rods: safety (SA), shim (SH), and regulating (RR). The primary function of safety rod is to shut-down the entire reactor in case of an accident. A shim rod is utilized to make coarse adjustments in neutron density. A regulating rod is used to make fine adjustments and power regulation. Beside them, there is a special control rod type in the reactor core. It is air-follower transient rod (TR) which is used for square wave pulse operation. In Figure 3-1, the fuel rods and four control rods are shown in detail. Outer diameter of each section in fuel and control elements (SA, SH, and RR) is tabulated in Table 3-1 and the outer diameter of each section in transient rod is tabulated in Table 3-2.

Table 3-1: Outer diameter of each section in fuel and fuel follower control rods (SA, SH and RR)

Fuel Rod	Outer Diameter (cm)	Control Rods (SA, SH, RR)	Outer Diameter (cm)
Zirconium Rod	0.44	Zirconium Rod	0.64
Fuel Meat	3.63	Fuel Meat	3.33
Graphite	3.63	Neutron Absorber	3.33
SS304 Clad	3.73	SS304 Clad	3.43

The pitch, which indicates the center-to-center distance between two adjacent fuel elements, is 4.318 cm (1.7 inches) in each hexagonal array. The PSBR core was first loaded with fresh 8.5 wt-percent U-ZrH_{1.6} fuel elements in 1965. Then, some fresh 12 wt-% fuel elements were added in 1972. Current PSBR core loading pattern at the time of MCNP calculations is called loading-53H in which 37 fuel rods are 12 wt-% and 65 fuel rods are 8.5 wt-%. This loading pattern went critical in May 2009 and core loading pattern is shown in Figure 3-2. In the core, the fuel temperature is monitored using three thermocouples installed inside instrumented fuel elements (I-11, I15, and I-16). The center thimble (CT) is located at the center of the core and it is surrounded by hexagonal rings. Starting from the centre thimble to the outward, the rings are designated as B, C, D, and so on.

Table 3-2: Outer diameter of each section in air follower Transient Rod (TR)

Control Rod (TR)	Outer Diameter (cm)
Air	3.33
Neutron Absorber	3.33
Al Clad	3.40

The PSBR has a movable core. The core is carried and supported by a suspension tower which is connected to a bridge on top of the pool. In the existing design, a large drum-shaped D₂O tank is utilized for neutron moderation and a graphite box which is located behind the D₂O tank is used for neutron reflection. When the reactor core is coupled with the D₂O tank, the neutron beam becomes available in the beam ports #4 and #7. An AutoCAD® [41] drawing of the reactor core with the aluminum tower, D₂O tank and graphite reflector is given in Figure 3-2.

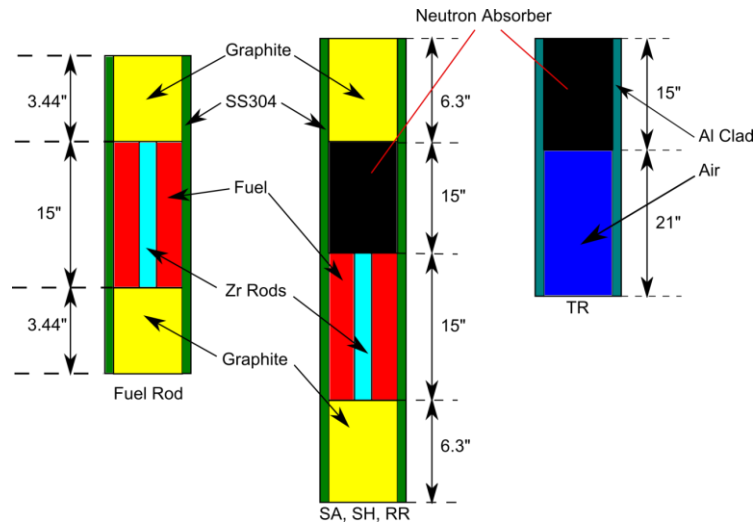


Figure 3-1: The length of each section in TRIGA fuel and control rods

Neutron flux was reported as $1 \times 10^{13} \text{ cm}^{-2} \text{ sec}^{-1}$ at the periphery of the core edge and $3 \times 10^{13} \text{ cm}^{-2} \text{ sec}^{-1}$ at the centre thimble for PSBR TRIGA Mark III reactor at 1 MW power [42]. Because of the large prompt negative feedback property of the TRIGA fuel design, the PSBR can be pulsed up to 2000 MW for 10 and 20 milliseconds.

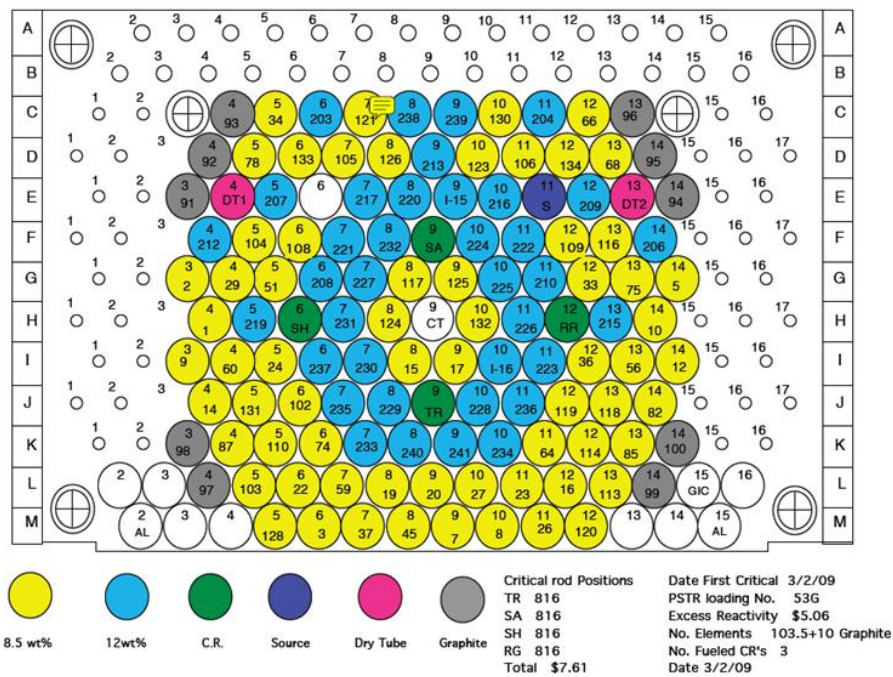


Figure 3-2: Core loading-53H (2009) schematic drawing

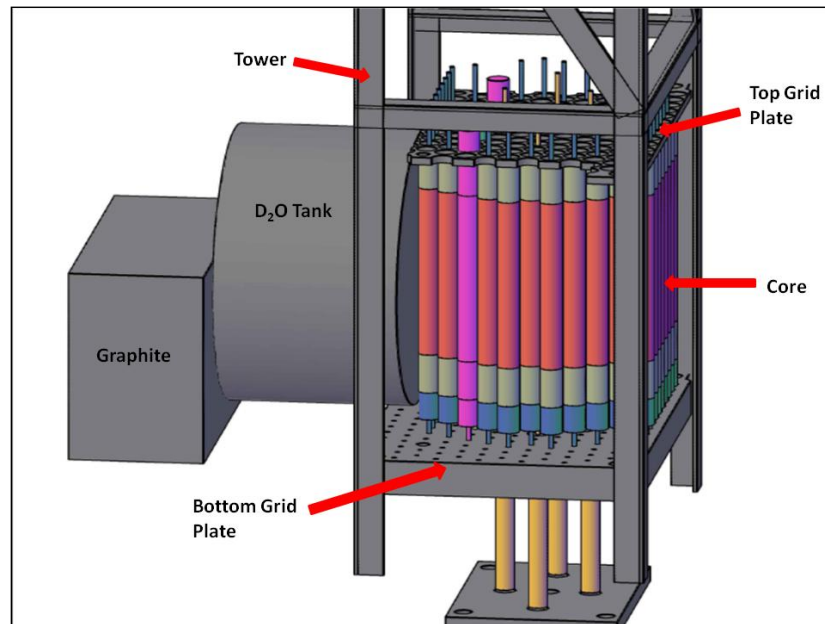


Figure 3-3: A 3D AutoCAD® drawing of the existing reactor core-moderator assembly with graphite reflector and tower

Existing PSBR Design Issues

The deficiencies of the existing design can be categorized into two parts: the geometrical deficiencies and the hydrogen gamma contamination problem. These two problems are related to each other and in the following subsections; they are explained in detail.

Geometrical Design Issues

Geometrical deficiencies originate from the design of the reactor core-moderator assembly, as well as the installation of the beam ports with respect to the core centerline. The PSBR was first designed with plate-type MTR fuel elements enriched up to 93%. The reactor was primarily build for analyzing the nuclear properties of materials, determining reactor dynamics,

and examining the effects of radiation on materials. Seven beam ports were constructed in the first design in order to fulfill this purpose.

During an upgrade in 1965, reactor was converted from plate-type fuel core to a TRIGA core. This conversion brought three major advantages to the facility. First, steady-state power was increased from 200 kW to 1 MW. A higher neutron flux was obtained to the experiment facilities from the new TRIGA core. Second, fuel enrichment need decreased from 93% to less than 20% and reactor was switched to a low enriched uranium safeguards category. Third, because of added security offered the inherent prompt negative feedback characteristics of the TRIGA fuel, pulsing capability was added to the core.

The 1965 conversion resulted in a partial loss of the experimental capability of the facility. The active length of plate-type MTR fuel elements are 24 inches. On the other hand, it is 15 inches in the TRIGA fuel elements. After the conversion, only one beam port stayed directed along the core center, four beam ports stayed five inches below the core center, and the remaining two stayed eleven inches below the core center. Therefore, six of the seven available beam ports are limited in neutron beam utilization for nuclear research and education. In Figure 3-4, existing geometrical configuration of the beam ports with respect to pool floor and the TRIGA fuel are plotted to better illustrate this problem. A 3D AutoCAD[®] drawing of the existing reactor core, D₂O tank, graphite, and seven beam ports extended toward the core are shown in Figure 3-5 [21]. Figure 3-6 gives a top view of the Figure 3-5. These figures demonstrate the geometrical alignment mismatch of the existing neutron beam ports with respect to the current core-moderator assembly.

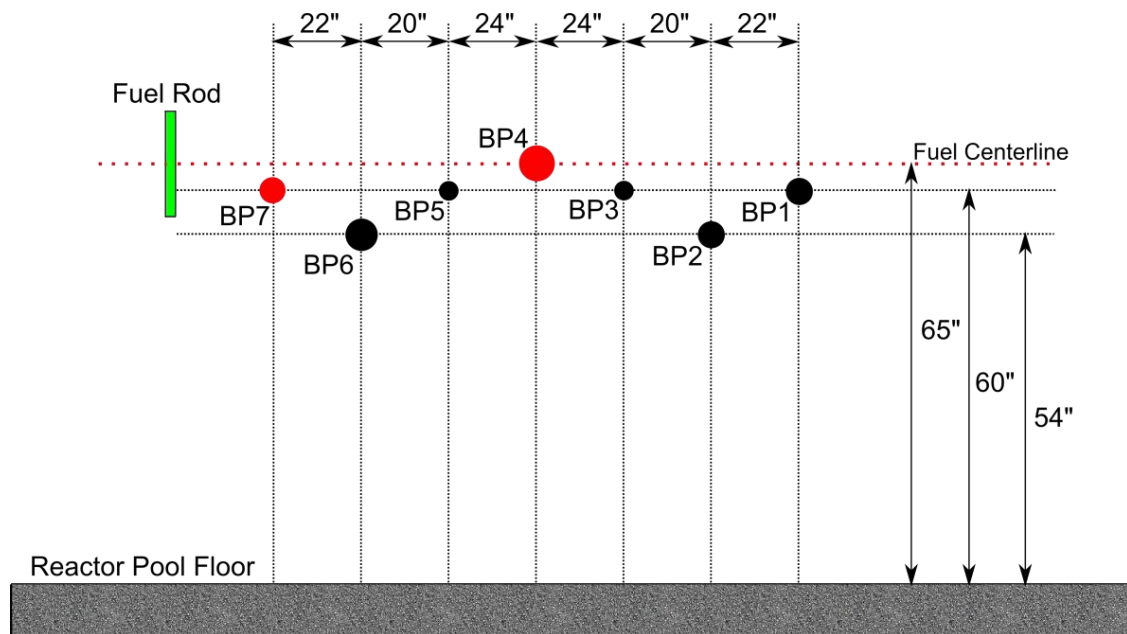


Figure 3-4: Relative locations of beam ports with respect to pool floor

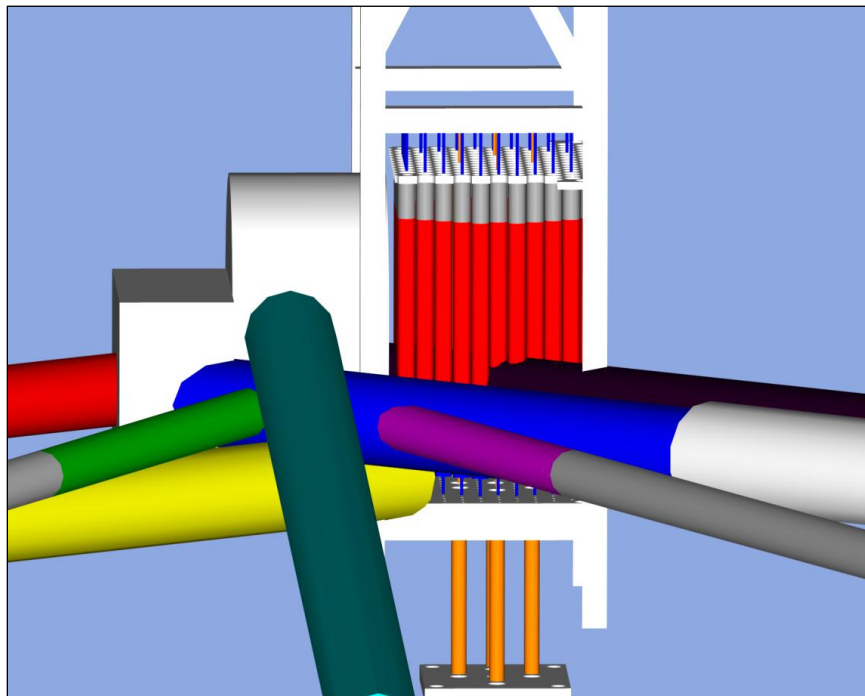


Figure 3-5: A 3D AutoCAD® drawing of the existing core-moderator assembly layout with extended views of existing beam ports (side view) [21]

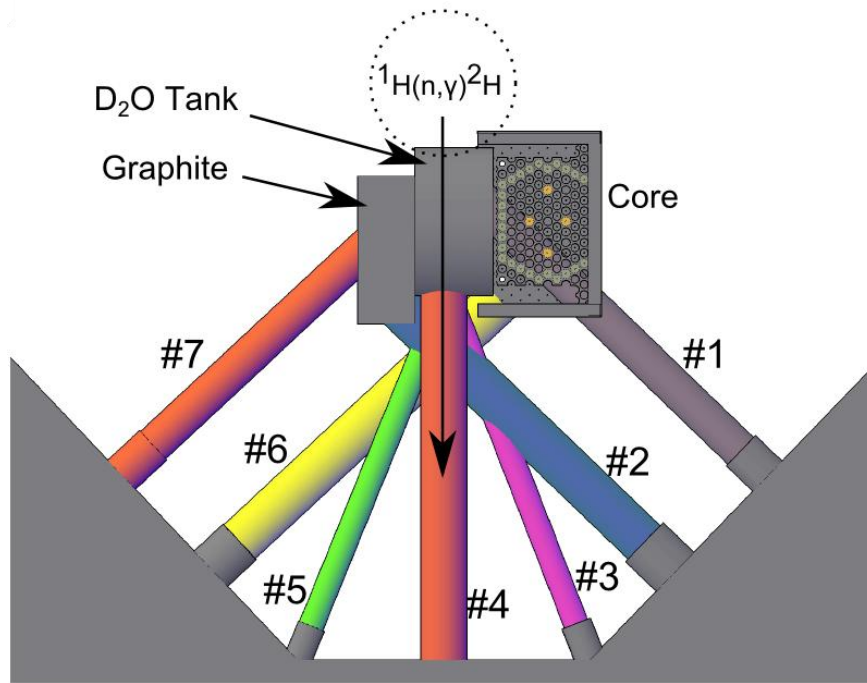


Figure 3-6: A 3D AutoCAD® drawing of the existing core-moderator assembly layout with extended views of existing beam ports (top view) [21]

Gamma Contamination of Neutron Beam in BP4

The second problem that plagues the neutronic performance of existing design is the contribution of high-energetic prompt gamma-rays to the neutron beam in BP4. These prompt gamma-rays are produced by the neutron capture of the hydrogen in water due to the ${}^1\text{H}(n,\gamma){}^2\text{H}$ reaction which mainly occurs at the sides of the D₂O tank as shown in Figure 3-6. As a result of this reaction, a significant amount of prompt gamma-rays with an approximate energy of 2.2 MeV can enter into BP4. The gamma contamination problem in this section was analyzed in detail. The benchmarking of the gamma contamination problem was achieved with MCNP simulations and experiments. The simulated and measured results were used for new beam port designs and comparison in the following chapters.

Benchmarking of the Gamma Contamination

The gamma contamination problem was previously observed during the Neutron Depth Profiling (NDP) measurements of an Intel-SEA2 borophosphosilicate glass (BPSG) sample at Penn State University Neutron Depth Profiling (PSU-NDP) facility [5]. The purpose of the experiment was to calculate the concentration vs. depth profile of Boron-10 isotope in BPSG sample. The BPSG is a calibrated sample in which boron atoms are doped as a p-type dopant. The abundance of Boron-10 in the sample is known as 19.9%. In NDP measurements, BPSG sample was bombarded by the neutron beam in BP4. As a result of neutron capture reaction by the Boron-10 atoms in the sample, two alpha particles and two lithium recoil atoms are released from the sample surface. The energy of each particle was collected by a silicon PIN photodiode detector and an energy spectrum was obtained at the end of NDP measurements. However, the detector effectively collected the prompt gamma-rays which currently exist in the neutron beam in BP4. Therefore, the energy spectrum had a very high background which shaded the other peaks up to 1 MeV. In order to find the source of this background in the energy spectrum, NDP measurements of BPSG were also made at National Institute of Standard Technology (NIST) Cold-NDP facility [43] since the neutron beam in this facility has a very low gamma component compared to the PSU-NDP facility. Thus, the energy spectrum collected by the BPSG sample did not have any background effect at NIST. Furthermore, PSU-NDP facility was modeled with Geant4 code with the same detector and sample material compositions and dimensions with the experiments [5]. In this model, the neutron beam was defined with Maxwell-Boltzmann energy spectrum without any gamma component. This was the only difference between the simulation and the measurement. Thus, the energy spectrum in Geant4 model represents the ideal experimental results that can be measured in PSU-NDP facility and it closely matches with the energy spectrum at NIST Cold-NDP facility. The comparison of the measured and simulated

energy spectra at PSU-NDP facility and NIST Cold-NDP facility showed that the only source of background in PSU-NDP facility was the gamma component of the neutron beam in BP4.

After benchmarking the problem, NDP facility was improved by installing a single-crystal sapphire crystal into the BP4. Then, the neutron and gamma beam in BP4 was calculated by measurements using activated gold foils and MCNP simulations before and after sapphire filter installation. The comparison of the results was used to estimate the neutron and gamma transmission rates in the sapphire filter.

MCNP Modeling and Measurement of the Neutronic Performance of Sapphire Filter in BP4

BP4 is utilized in almost all of the experimental facilities including neutron radiography, radioscopy, neutron depth profiling, neutron imaging, and slow neutron chopper spectroscopy measurements. Figure 3-7 shows a schematic layout of the PSBR core with loading pattern (53H), D₂O tank, graphite block and BP4. As seen in this figure, BP4 is looking to the D₂O tank. The beam port is divided into 4 sections, the two sections at the pool side are made of aluminum and the remaining two sections in the wall are made of steel. The diameters of these sections vary from 15.2 cm close to D₂O tank to 19.1 cm at the exit of the beam port. The total length of the beam port is 340 cm. BP4 contains a collimation section which consists of a bismuth crystal, a borated-aluminum (boral) aperture, and a lead plug in the first pipe close to D₂O tank. Sapphire filter was installed just after the lead plug as shown in the Figure 3-7. Two collimator pieces are plugged at the end of the last section to reduce the neutron beam diameter to 1.5 cm as well as to direct neutron beam to the experimental facilities.

Sapphire filter in the beam port is a hemlite-grade single crystal with 10.1-cm diameter and 7.62-cm thickness. It is one of the best filter materials available to mitigate the gamma-rays and fast neutrons [44]. The microstructure of sapphire filter contains super optical quality crystals

whose [001] axis is parallel to the incoming neutron beam. Also its transmission properties do not change in a constant irradiation field such as a beam tube of a reactor even after several years. Additionally cooling of sapphire filter to liquid nitrogen temperature is not required. In the following sections, the effects of the sapphire filter to the neutron and gamma beams in BP4 were analyzed by experiments and MCNP simulations.

Thermal and Epithermal Neutron Flux Measurements in BP4

The total thermal neutron flux and total epithermal neutron flux were measured in BP4 using bare (99.99%) and cadmium (Cd)-covered gold foils. Five bare and five cadmium covered gold foils were symmetrically placed on an aluminum (Al) holder. These measurements were performed by placing the Al holder to three different beam port locations, which are numbered as location #1, #2 and #3 in Figure 3-7. At location #1 (inner), foils were irradiated at 100 kW for 1 hr. Then, they were irradiated at 200 kW for 2 hours at location #2 (middle) and at 300 kW for 3 hours at location #3 (outer). After a cooling period of ~2 days, the activity of gold foils was measured at NAA laboratory using a sample changer and a high purity germanium (HpGe) detector. Results of these measurements were normalized to 1 MW power operation and are tabulated in Table 3-2. They were also corrected by using thermal and resonance self-shielding factors by considering the thickness of the gold foils employed in the measurements [45].

After sapphire filter installation, a set of gold foils were irradiated again at the same beam port locations and reactor power. The activity of each foil was then measured at NAA laboratory and total thermal and total epithermal neutron flux were calculated similar to the previous case. Results of these measurements are tabulated in Table 3-4. The percent decrease in total thermal neutron flux and total epithermal neutron flux at three different locations in BP4 were calculated by comparing the measured results with and without sapphire filter given in Tables 3-3 and 3-4

and the results are given in Table 3-5. The results given in this table are significantly different from each other. Approximately 52% of epithermal neutrons were filtered from the neutron beam with a 20% reduction in total thermal neutron flux at location #3. The neutrons moving along the beam port centerline (dotted line in Figure 3-7) can reach to collimator exit. Therefore, sapphire filter are only filtering these neutrons that can contribute to the neutron beam at location #3. On the other hand, the sapphire filter is prominently filtering the neutrons reaching to measurement locations #1 and #2 in any direction. That is why the change in the thermal and epithermal neutron flux is higher at these locations compared to location # 3 after sapphire filter installation.

Table 3-3: Measured total thermal and total epithermal neutron flux at three measurement locations in BP4 before sapphire filter installation

Measurement Locations	Total Thermal Neutron Flux (n/cm²/sec)	Total Epithermal Neutron Flux (n/cm²/sec)
Inner (1)	$1.71 \times 10^{10} \pm 6.20 \times 10^7$	$3.40 \times 10^8 \pm 1.20 \times 10^6$
Middle (2)	$2.03 \times 10^8 \pm 1.38 \times 10^6$	$3.29 \times 10^6 \pm 1.76 \times 10^4$
Outer (3)	$4.23 \times 10^7 \pm 4.33 \times 10^5$	$1.52 \times 10^6 \pm 1.01 \times 10^4$

Table 3-4: Measured total thermal and total epithermal neutron flux at three measurement locations in BP4 after sapphire filter installation

Measurement Locations	Total Thermal Neutron Flux (n/cm²-sec)	Total Epithermal Neutron Flux (n/cm²-sec)
Inner (1)	$1.06 \times 10^{10} \pm 1.20 \times 10^8$	$1.40 \times 10^8 \pm 1.38 \times 10^6$
Middle (2)	$1.50 \times 10^8 \pm 1.08 \times 10^6$	$1.33 \times 10^6 \pm 1.20 \times 10^4$
Outer (3)	$3.39 \times 10^7 \pm 2.71 \times 10^5$	$7.33 \times 10^5 \pm 4.41 \times 10^3$

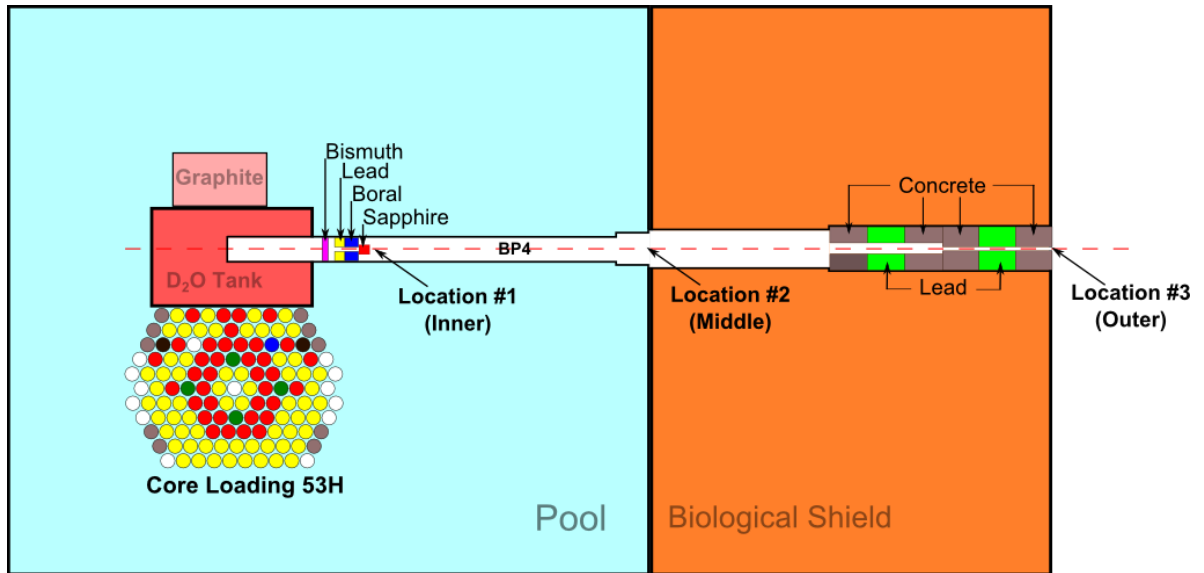


Figure 3-7: A schematic drawing of the existing reactor including core loading-53H, D₂O tank, graphite reflector, and BP4. The measurement locations are labeled as location #1, #2 and #3.

Table 3-5: The percent decrease in total thermal neutron flux and total epithermal neutron flux at the measurement locations in BP4 after the sapphire filter installation

Measurement Locations	Decrease in Total Thermal Neutron Flux (%)	Decrease in Total Epithermal Neutron Flux (%)
Inner (#1)	38.01 ± 1.18	58.83 ± 1.05
Middle (#2)	26.11 ± 0.99	59.73 ± 1.03
Outer (3)	19.92 ± 1.30	51.81 ± 1.05

Gamma Dose Measurements at Neutron Beam Laboratory

The gamma dose measurements were performed using Ludlum 375/2 Geiger Muller (GM) tubes with 1-33-2 GM probes. Measurements were performed at different reactor power

levels, with and without sapphire filter in at the exit of BP4 (location #3). Results are given in Table 3-6. Initial measurements were conducted while the beam shutter was closed to calculate the background in the neutron beam laboratory.

By comparing the measured gamma-rays with and without the sapphire filter in BP4, the gamma attenuation rate in the sapphire filter was estimated as ~40% from the measurements. The relative error in these calculations was assumed to be 15% following the manufacturer specifications for the GM tube. However, the presence of the Neutron Imaging System (NIS) [46] at the neutron beam laboratory during these measurements significantly affected the accuracy of the results. The NIS system contains a mirror which is located along the primary neutron beam direction. It is predicted that some of the gamma-rays may be backscattered from this mirror and contribute to the gamma dose counts in the GM tubes. Therefore, the measured results may overestimate the actual result. Since it was not possible to move the NIS system from the beam laboratory, we were not able to repeat the experiments.

Table 3-6: Gamma dose measured at neutron beam laboratory for different reactor power operations

1.5" Collimator installed in BP4	Power [kW]	Door Shut (background)	Door Open
		Dose Rate (mR/hr)	Dose Rate (mR/hr)
Sapphire Filter Out	50	2	470
	100	6	950
	150	9	1430
	800	36	8200
Sapphire Filter In	100	3	630
	200	8	1250
	800	26	5020

MCNP5 Simulation of Neutron and Gamma Flux in BP4

In this section, an MCNP5 (v1.6) simulation was performed to estimate the neutron flux and gamma dose at the BP4 exit. During the simulation, an advanced calculation methodology was implemented to reduce the computation time and the relative error in the flux tallies [47]. This approach was first employed by Sarikaya and Alim [19]. In this technique, MCNP simulations were performed in two steps using two special cards: SSW (surface source write) and SSR (surface source read) cards. In the first step, reactor core was modeled with D₂O tank, graphite reflector, and a portion of BP4. This is called “*full core*” model shown in Figure 3-8. In this model, core loading 53H was modeled with the fuel compositions, which were calculated by TRIGSIMS (TRIGA Simulator version S) code [15]. Then, a criticality calculation defined by KCODE card of MCNP was performed with 700 histories and 1 million particles per history in active cycles. MCNP results were estimated with a relative error of less than 1%. In the full core model, fission neutrons and neutron induced gamma-rays incoming into the bismuth filter in any direction were written to a file named *surface source file* using the SSW card. This file records all the necessary information of the particles entering into the bismuth filter; as such energy, position, direction, and weight of the simulated particles.

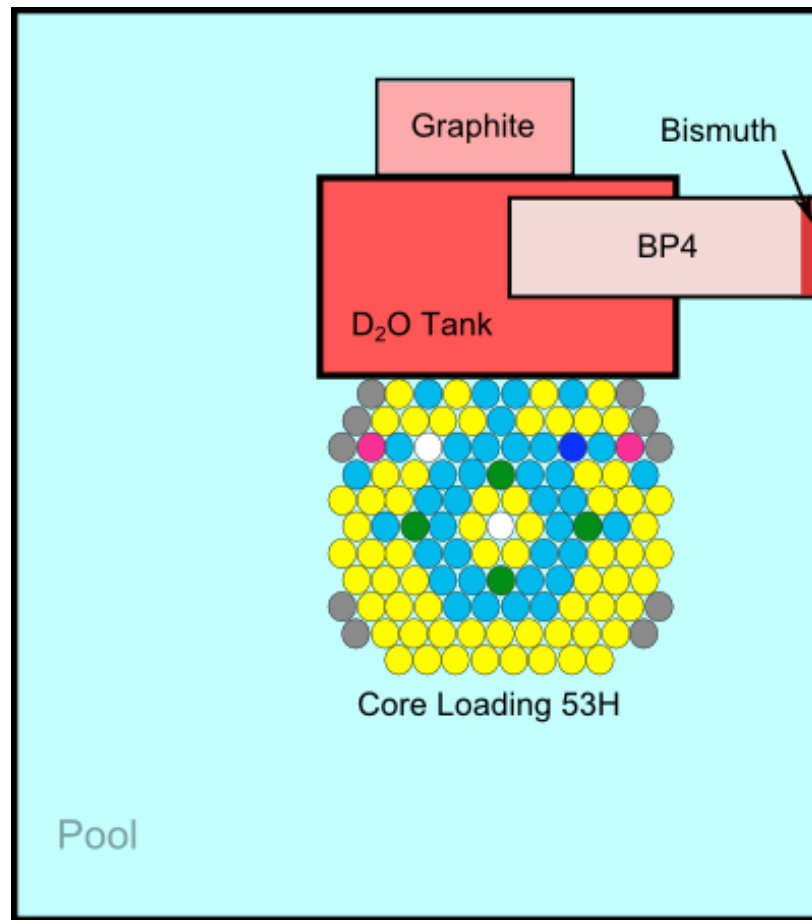


Figure 3-8: Full core model in the existing PSBR design

In the second step, MCNP simulation was achieved by modeling only BP4. Thus, the second model is named as “BP4” model shown in Figure 3-9. The source in this model is generated by reading the *surface source file* using SSR card. Then, the neutron and gamma flux spectra were calculated at three measurement locations in BP4. This calculation approach brought two major advantages: (1) criticality (KCODE) calculation was converted into a fixed source problem with less geometrical detail and (2) computational time was significantly reduced. Additionally, the number of source particles reaching to the tally locations in BP4 was extensively increased by oversampling the recorded particles in the surface source file.

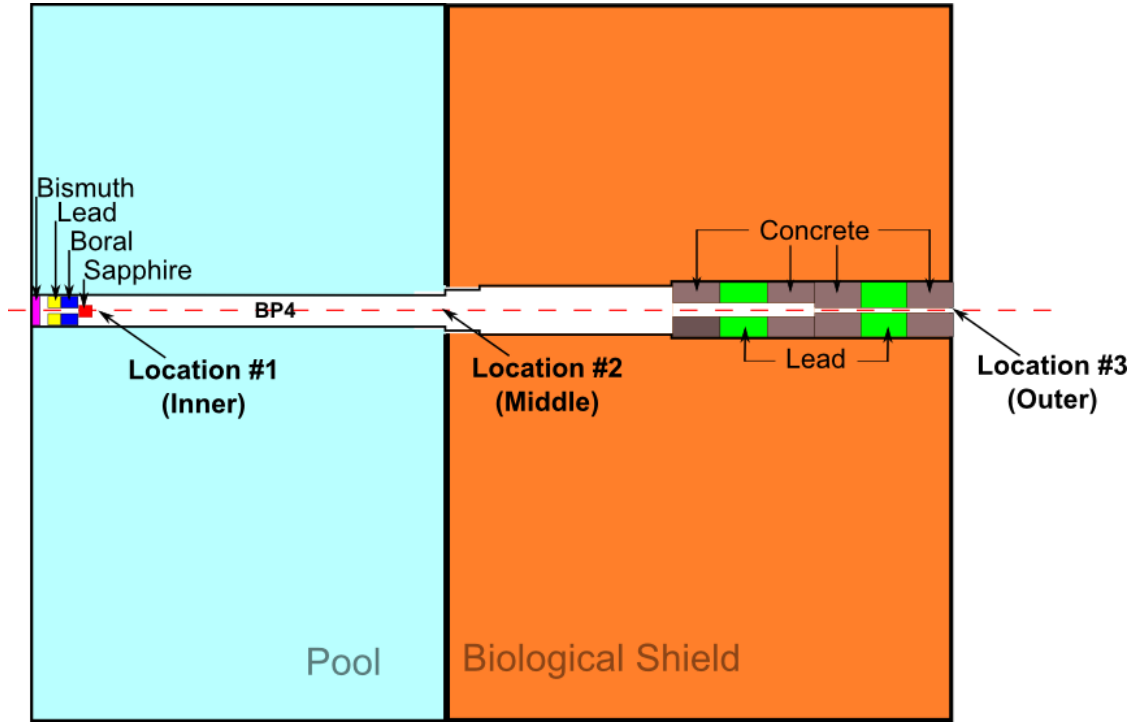


Figure 3-9: BP4 model in existing PSBR design

The MCNP calculated thermal flux is the integral form of the thermal flux defined in Equation (3-1). However, the measured thermal neutron flux is actually a 2200 meter-per-second flux at 20 °C, which corresponds to a kinetic energy of 0.0253 eV. A proper conversion is necessary in order to compare the simulation results with the measurements. The integral form of neutron flux is defined as

$$\phi_T = \int \phi(E)dE \quad (3-1)$$

The neutron flux spectrum in BP4 follows the Maxwell-Boltzmann energy and velocity distribution [38]. Therefore, integral form of the thermal neutron flux can be written as follows.

$$\phi_T = nv_{avg} = \frac{2}{\sqrt{\pi}}nv_T. \quad (3-2)$$

In Equation (3-2), v_{avg} is the average neutron velocity, v_T is the thermal neutron energy and n is the thermal neutron density. The relationship between average neutron velocity and the thermal neutron velocity is

$$v_{avg} = \frac{2}{\sqrt{\pi}} v_T. \quad (3-3)$$

The thermal neutron velocity, v_T is the most probable velocity at temperature, T . It is related to the most probable velocity at room temperature as follows.

$$v_T = \sqrt{\frac{2kT}{m}} = v_0 \sqrt{\frac{T}{T_0}}. \quad (3-4)$$

The 2200 meter-per-second flux, ϕ_0 is expressed as

$$\phi_0 = nv_0. \quad (3-5)$$

ϕ_0 is related to the integral form of flux (MCNP) with the following relation.

$$\frac{\phi_0}{\phi_T} = \frac{nv_0}{\frac{2}{\sqrt{\pi}}nv_t} = \frac{\sqrt{\pi}}{2} \frac{v_0}{v_T} = \frac{\sqrt{\pi}}{2} \sqrt{\frac{T_0}{T_{abs}}}, \quad (3-6)$$

where

$$T_0 = \frac{\frac{1}{2}mv_0^2}{k} = \frac{\frac{1}{2}(1.67492 \times 10^{-27} \text{ kg}) \left(2200 \frac{\text{m}}{\text{s}}\right)^2}{1.38066 \times 10^{-23} \text{ J/K}} = 293.60 \text{ K}. \quad (3-7)$$

The absolute temperature of the medium, T_{abs} in BP4 is 25 °C (298.15 °K). Then, the relation between ϕ_0 and ϕ_T can be written as

$$\phi_0 = \phi_T \frac{\sqrt{\pi}}{2} \sqrt{\frac{293.60}{298.15}}. \quad (3-8)$$

Total thermal and total epithermal neutron flux estimated by MCNP simulations at three beam port locations are tabulated in Tables 3-7 and 3-8. First the measured results were converted to integral form of the flux using Equation (3-8) and were then given in these tables for

comparison. Additionally, MCNP predicted neutron flux spectra at the three tally locations are shown in Figure 3-10.

As seen from the results, the total thermal neutron flux in BP4 was predicted with a relative error less of than 2% by the MCNP simulation in the case of no sapphire filter installation. However, the measured total epithermal neutron flux is significantly different than MCNP results. It is predicted that the main cause of this discrepancy is deficiencies of the nuclear data in the epithermal energy region [48]. Use of a newer cross section data set, as such ENDF/B-VII.1, may correct this problem. Compared to former data sets, notable improvements in the MCNP results for benchmark test problems were reported in reference [49].

Table 3-7: Comparison of the total thermal neutron flux between measurements and MCNP simulation before sapphire filter installation

Measurement Locations	<u>Experiment</u> Total Thermal Neutron Flux (n/cm²-sec)	<u>MCNP</u> Total Thermal Neutron Flux (n/cm²-sec)
Inner (1)	$1.75 \times 10^{10} \pm 3.47 \times 10^8$	$1.79 \times 10^{10} \pm 9.20 \times 10^7$
Middle (2)	$2.00 \times 10^8 \pm 2.10 \times 10^6$	$2.07 \times 10^8 \pm 3.40 \times 10^6$
Outer (3)	$4.23 \times 10^7 \pm 7.70 \times 10^5$	$4.36 \times 10^7 \pm 7.00 \times 10^5$

MCNP simulation was not successful to calculate the neutron flux spectra after sapphire filter because of the lack of cross section data set for sapphire filter in the evaluated data tables. Instead of MCNP simulation, a Monte Carlo analysis was performed by employing a measured cross section data for the 7.2cm-thick hemlite-grade sapphire crystal. This data was obtained by the neutron transmission measurement with the sapphire filter at NIST reactor [50]. Hence, the data obtained at NIST can be treated as the transmission probability of neutrons in any energy in

the sapphire filter. Every neutrons interacting in the sapphire filter was assumed to be absorbed by ignoring the scattering probability.

Table 3-8: Comparison of total epithermal neutron flux between measurements and MCNP simulation before sapphire filter installation

Measurement Locations	<u>Experiment</u> Total Epithermal Neutron Flux (n/cm²-sec)	<u>MCNP</u> Total Epithermal Neutron Flux (n/cm²-sec)
Inner (1)	$3.39 \times 10^8 \pm 7.33 \times 10^6$	$5.55 \times 10^8 \pm 4.00 \times 10^6$
Middle (2)	$4.23 \times 10^6 \pm 9.33 \times 10^4$	$5.30 \times 10^6 \pm 2.27 \times 10^5$
Outer (3)	$1.43 \times 10^6 \pm 2.80 \times 10^4$	$1.36 \times 10^6 \pm 1.46 \times 10^5$

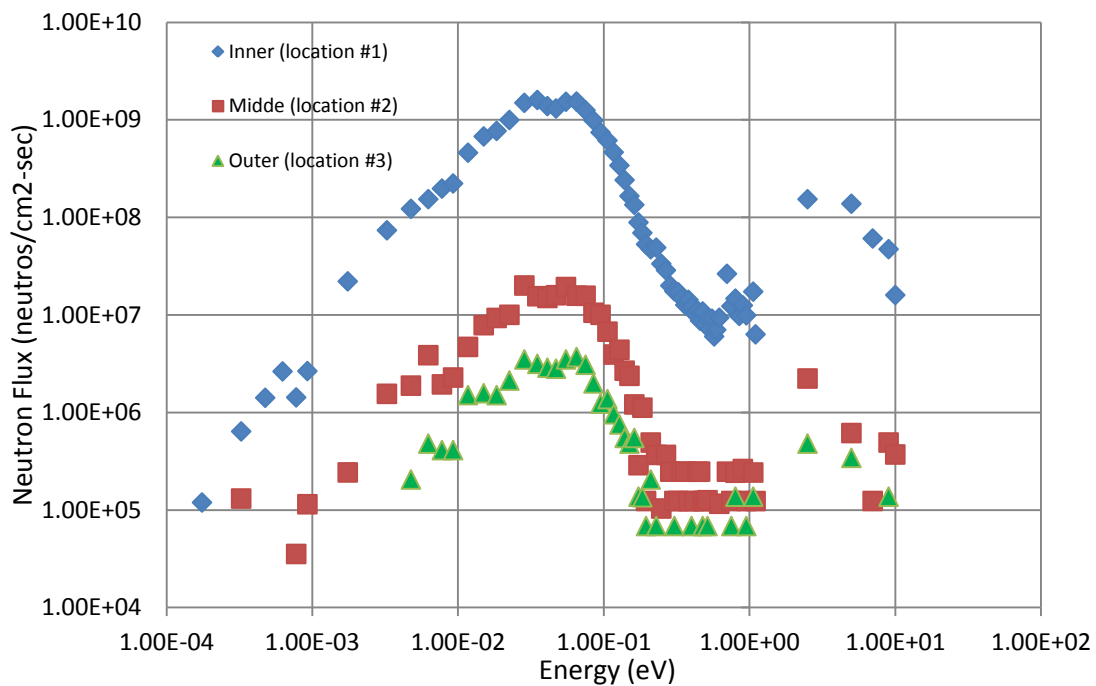


Figure 3-10: Comparison of neutron flux spectra calculated by MCNP simulations at three measurement locations in BP4 for 1 MW reactor power operation

By using the measured transmission cross section data, the decrease in the total thermal neutron flux and in the total epithermal neutron flux were calculated and sub-sequentially then compared with experimental results at the inner (location #1) and outer (location #3) of BP4. The results are tabulated in Table 3-9. Monte Carlo simulation results are in very close proximity of the measured values. There is a ~10% difference calculated between the measurements and simulation results. The reason of the difference is mainly caused by the Monte Carlo simulation approach in which the elastic scattering is ignored in the sapphire filter and the measured cross section data was employed to define the neutron loss probability in sapphire filter.

The attenuation rate of the fast neutron flux in the 7.62-cm thick sapphire filter was estimated using the free atom cross section of Al_2O_3 which is 0.33 cm^{-1} [44]. It was calculated using the following formula.

$$\text{Attenuation Rate (\%)} = 1 - \exp(-\Sigma t). \quad (3-9)$$

where Σ is the macroscopic cross section (0.33 cm^{-1}) of Al_2O_3 and t is the thickness (7.62 cm) of the filter. As a result of this calculation, the fast neutron attenuation rate was predicted as 92% in the hemlite-grade sapphire crystal.

The gamma flux spectra at the end of BP4 (location #3) was predicted by MCNP simulations before and after the sapphire filter installation. The MCNP predicted gamma flux spectra at location #3 are shown in Figure 3-11. The total gamma flux was then calculated by integrating the area under each gamma flux spectra. The comparison of total gamma flux before and after sapphire filter showed that majority of the gamma-rays will be eliminated from the neutron beam in BP4 after sapphire filter installation. The gamma attenuation rate was predicted as $60.3\% \pm 1.3\%$ by the MCNP simulation. However, this is significantly higher than the measured value of 40% because of the high background created by the neutron imaging system during the gamma-dose measurements.

As a summary, ~60% reduction in the total gamma dose and 50% reduction of the total epithermal neutron flux were observed at the end of BP4 (location #3) after the hemlite-grade sapphire crystal installation. The decrease in total thermal neutron flux was only ~20% compared to the original BP4. The predicted fast neutron attenuation rate in the sapphire filter is 92% calculated by using the free atom cross section of Al_2O_3 in the fast region. Additionally, the 2.2 MeV prompt gamma-ray peak in the gamma spectra was diminished by a factor of 3 in the sapphire filter.

MCNP simulation of the existing PSBR design gave reasonably accurate results in BP4 compared to the measurements. Another important outcome of the analysis is the success of the MCNP simulation approach performed in two steps with SSR and SSW cards. The only limitation faced in the MCNP calculations is the lack of cross section data set available for the hemlite-grade sapphire filter. Although Monte Carlo simulation approach is fairly accurate, a different cross section data set was used in the neutronic analysis of new beam ports #1 and #2 in which sapphire filter is installed. The details of this analysis can be found in Chapter 7.

Table 3-9: Comparison of thermal and epithermal neutron transmission rates at the end of BP4 between Monte Carlo simulation and measurements

Measurement Location	Decrease in Thermal Neutron Flux (%) (Experiment)	Decrease in Thermal Neutron Flux (%) (Monte Carlo)	Relative Error (%)
Inner (#1)	38.10	30.58	10.70
Outer (#3)	19.92	11.25	10.20
Measurement Location	Decrease in Epithermal Neutron Flux (%) (Experiment)	Decrease in Epithermal Neutron Flux (%) (Monte Carlo)	Relative Error (%)
Inner (#1)	58.83	65.54	10.24
Outer (#3)	51.81	57.20	10.40

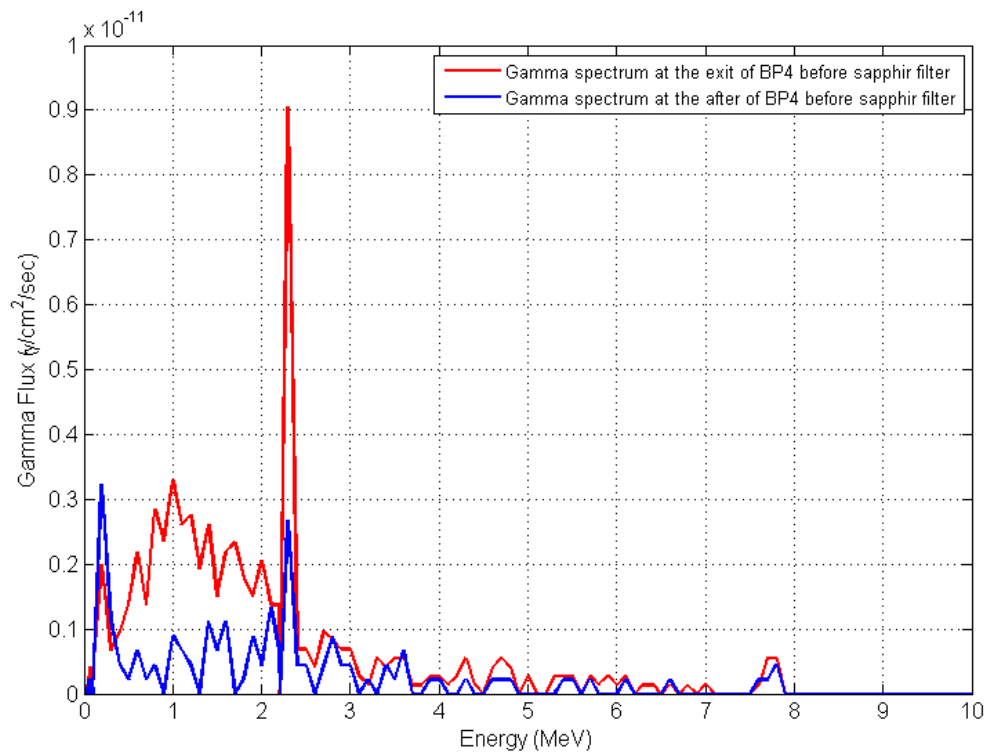


Figure 3-11: MCNP predicted gamma flux spectra at the end of BP4 (location #3) before and after sapphire filter installation

Utilization of Existing Beam Hall

The Penn State Breazeale Reactor (PSBR) went critical for the first time on 1955 as a part of “Atom for Peace” program. The reactor has been an international training and education center. The Breazeale Reactor was primarily build for analyzing nuclear properties of materials, determining reactor dynamics, and examining the effects of radiation on materials. Seven beam ports were constructed in the first design in order to accomplish this purpose. After the conversion from MTR type fuel to TRIGA fuel, usage of six beam ports was severely hindered.

In the current neutron beam hall, neutron imaging (NI), instrumental neutron activation analysis (NAA), neutron depth profiling (NDP), soft error rate measurements and detector testing processes are performed in BP4, and neutron transmission measurements are conducted in BP7.

Therefore, the currently present gamma contamination problem in BP4 seriously affects almost all of measurements conducted in the current beam port facilities. Although the 7.6-cm thick hemlite-grade sapphire filter installation eliminated some of the gamma-rays from the neutron beam in BP4, further improvements are necessary in order to advance the quality and capability of the facility.

Chapter 4

A NEW CORE-MODERATOR ASSEMBLY DESIGN for the PSBR

In this chapter, a new core-moderator assembly design is proposed for the PSBR. The purpose of a system design is to find solutions to the problems based on the requirements of the system. This is why the first section gives a brief explanation of the design methodology. The deficiencies of the existing PSBR design were introduced and the requirements to expand the neutron beam laboratory were identified in Chapter 3. Now, a proposed new core-moderator design is introduced based on these system requirements.

Design Methodology

The design of a core-moderator assembly and neutron beam port for a research reactor is a very complicated problem and requires special methods. A research reactor has several mechanical components working concurrently and also independently. Every component can be considered as a functional building block. Every block has different behavior and purpose. The reactor core is the hearth of the entire system. It contains all the fuel elements, wherein the fission chain reaction takes place and neutrons are generated. The moderator bears the function to slow-down the fission neutrons in the thermal and sub-thermal energy region. Another critical component, beam ports; are utilized to transport and direct the thermalized neutrons to the experimental facilities. In a reactor system, all of these components or functional blocks are connected and related to each other. Therefore, design requirements should be carefully selected by considering the behavior of each functional block.

The “Decomposition and Representation” approach is one of the design methods to reformulate a problem into small scale sub-problems [51][52]. It simplifies the complex engineering design problems in order to make easy design decisions. Decomposition methodology has been successfully applied to many complex engineering problems in various scientific and industrial fields [53][54]. In mechanical engineering, decomposition can be applied to three design problems: *product decomposition*, *process decomposition*, and *problem decomposition*. Product decomposition is generally implemented to physical elements of the product. It separates a product into physical components. On the other hand, process decomposition focuses on the entire design process independent of the product. This approach is applied to problems in which flow of elements or information exists. Problem decomposition is the basis of the design optimization and it was implemented to this analysis. Several different design approaches have been proposed to decompose complex problems ranging from classical problems to artificial intelligence in problem decomposition.

The primary design approach used in this thesis is the decomposition of overall design requirement into sub-requirements similar to problem decomposition. In the following sections, this approach is explained in detail.

Design Requirements

Design requirements define the desired solution of a problem. It is vital to clearly express a well-formulated design goal for the successful completion of the design process [55]. Because of the complexity of the systems design as a whole, the design process need to be decomposed into separate stages by defining sub-requirements. Once defined sub-goals are attained, the overall design requirement can be accomplished.

In this study, decomposition approach was applied by identifying the sub-requirements of the design process. The main objective or the overall design requirement of this study is to redesign the PSBR core, moderator, along with beam ports which will provide maximum thermal neutron flux with scarce fast neutron flux and gamma-dose to the experimental facilities. Using this approach, the number of simultaneously utilized neutron beam ports will be increased and hence the experimental capability of the reactor will be advanced. In order to achieve this purpose, the deficiencies of the existing design were determined first in the previous chapter. It was revealed that the existing core-moderator assembly suffers from a reduced number of simultaneously employable neutron beam ports in the beam hall. Moreover, the current core-moderator assembly also increases the number of hydrogen gamma-rays in the neutron beam in BP4. Therefore, the first step in the design process is the redesign of the core-moderator assembly that will allow the simultaneous use of five new beam ports in the facility in a manner that will eliminate the hydrogen gamma beam in the foretold beam ports. After this is accomplished, the second step will be the proper installation of these beam ports to the, now newly designed, core-moderator assembly. Proceeding from that, the final design goal will be the expansion of the beam hall with several experimental facilities.

By considering all these factors, design requirements of this study can be summarized in steps as follows.

- Design of a new PSBR core-moderator assembly
- Determination of a new moderator shape which can be coupled with the new core
- Change of the tower design to support the new core, hence the top and bottom grid plates
- Installation and coupling of new neutron beam ports to the new core-moderator assembly

Design Constraints

Design constraints define the limitation on the conditions under which a system is designed. It is important to identify present design constraints to as they aid in defining an overall boundary to the design process. The major constraints to design of the PSBR core-moderator assembly are mainly due to the geometrical factors. Available floor space in the building, unmovable structures in the floor plan, reactor core and moderator configuration, and the beam ports are the main geometric constraints.

Since the design process was subdivided into separate stages, the design constraints of each one needs to be identified. The first step is the coupling of the reactor core and moderator assembly. There are four main constraints for this step:

- Excess reactivity of the reactor core
- The tower design to support the new top and bottom grid plates
- Gamma contamination problem in the beam port facilities
- Reduction of cross-flow rate into the core due to the new moderator shape

In the new design, the current core loading pattern is going to be implemented with the existing fuel and control rods. However, the graphite rods will be removed from the core to promote coupling between the core and moderator. This loading change may result a sudden increase in the reactivity loss of the system. To mitigate this adverse effect, a new moderator tank is proposed to be installed that will span more than half of the proposed core. This will provide an excess reactivity gain in the core. These two competing effects will be analyzed by performing a criticality calculation using MCNP code. As mentioned in the previous chapter, the gamma contamination problem is significantly hampering the experimental capability of the facility. Therefore, one target of the new core-moderator assembly should be the reduction of the gamma-rays in the new beam port facilities. Additionally, the cross-flow entering into the reactor core is

affected by the new moderator shape. As mentioned in Chapter 2, the existing core is mainly cooled by cross-flow. The reduction in the cross-flow may result a significant temperature increase in the fuel rods. The concern is that this increase shall not compromise the PSBR safety limits which are defined in terms of maximum fuel temperature of 1150 °C. Therefore, flow and temperature profiles in the new core-moderator assembly will be analyzed to verify the safety of the design.

The design constraints for the second design step, which is the installation of neutron beam ports to the core-moderator assembly, are mainly about the available infrastructure; namely floor space in the beam port facility and neutron economy. In the new design, it is expected to utilize each beam port for an experimental instrument in the new beam hall. Therefore, the utilization of neutron beam in each beam port must exceed the minimum required neutron flux levels to satisfy proper functioning of the instruments. The location of each instrument in the neutron beam hall will be chosen based on the logistics of available floor space.

New Core and Moderator Configuration

In this section, a new core-moderator assembly design was introduced. The design process was started by analyzing the existing core and moderator designs. New assembly design was chosen based on the design constraints specified in the previous section. 3D AutoCAD® drawings of the existing reactor core and core-moderator assembly were previously given in Figures 3-3 and 3-4, respectively. The existing reactor core is coupled with the D₂O tank from the front face. The remaining faces of the reactor core are exposed to the pool and hydrogen gamma-rays are mainly generated at these locations. Therefore, either the core or the moderator shape is a key issue and needs to be modified in order to prevent the hydrogen gamma-ray generation in the vicinity of the core-moderator interface. Another key issue is the number of available beam ports,

which can be simultaneously utilized in the new PSBR design. The new moderator shape will need to maximize this as well.

The most important factor in the decision-making processes in this study is the calculation of the optimum moderator tank shape. Since the calculated shape directly affects the cores' and beam ports' design, computer based simulations are necessary to evaluate the optimum moderator shape; since it is not possible to build and evaluate the new design via physical experiments.

Using MOZAIK code, an optimum moderator shape was calculated in the existing core design and later in a new design in which existing BP4 was modeled along the core centerline [15]. The optimum moderator shape calculated by this code was similar to the existing D₂O tank. Although the calculated moderator shape does provide a maximum thermal neutron flux in the beam port facility, it does not consider prompt gamma contamination in the new beam ports. In its current state, it is also not capable of calculating an optimum moderator shape for a core design where five new beam ports are utilized. MOZAIK code was not used in this study because of its limited capability and thus a novel different calculation methodology was followed aiding in making of design decisions.

Some early design decisions for the optimum moderator shape were previously proposed in Butler's thesis [21] by analyzing the other TRIGA reactor designs. Butler suggested modifying the moderator shape, while preserving the core and tower designs. Three different moderator shapes were suggested: Horseshoe, crescent, and rectangular. These moderator shapes in the existing core design are illustrated in Figure 4-1. An apparent water gap between the suggested moderator and the reactor core are also portrait in this figure. This gap, however, should be filled by additional moderator materials in order to prevent neutron loss and prompt gamma-ray generation.

Crescent-shaped Moderator

The first investigated candidate shape is the crescent, as it allows the simultaneous utilization of five new beam ports in the facility. However, it presents a drawback; the water gap is the highest in such a core-moderator assembly. If this moderator tank is installed as suggested, the pool water in the core-moderator assembly will not only decrease the thermal neutron flux but will also give rise to hydrogen gamma-rays adversely affecting the new beam port facilities. Therefore, an alternative coupling of the moderator shape with the core has to be considered.

Horseshoe-shaped Moderator

By utilizing horseshoe-shaped moderator tank, five new beam ports can simultaneously be used and the water gap between core and this moderator tank is reduced with respect to a crescent-shaped one. Thus, the required additional material to fill the gap between core and moderator is reduced in a horseshoe shaped moderator. However, the only disadvantage of this design is that it is not easy to weld the new beam ports to the surfaces of this moderator shape.

Rectangular-shaped Moderator

Rectangular moderator tank is basically the same design with existing D₂O tank. The drawbacks of this particular design have been presented before, as such is not considered in the new design.

By considering the number of available beam ports, the ease of coupling the moderator tank with the core and the ease of installation of the beam ports, the best candidate in the new PSBR design is the crescent-shaped moderator tank. However, there will be a major drawback of

this moderator design. The core-moderator assembly cannot fully couple with this moderator shape when using the existing core and tower designs. A drawing of the existing reactor core with top and bottom grid plates is shown in Figure 4-1. The bottom grid plate is supported by connecting this grid plate to the tower from both ends. However, top grid plate is supported by connecting it to the tower from the back corners and to the bottom grid using the grid support stanchion (bar) from the front corners as shown in this figure.

As such a crescent shaped moderator tank cannot be coupled with the existing reactor core and tower. Because of the two front grid support stanchions, a crescent-shaped moderator tank cannot be placed in close proximity to fuel elements. The front stanchions are the cause of the unavoidable water gap between core and moderator. These two front grid support stanchions are shown in Figure 4-2. Therefore, both core and tower designs will have to be changed in order to eliminate this discrepancy in the design process. In the following sections, the progress to design a new core-moderator assembly is given step-by-step. In the first step, new top and bottom grid plates are introduced. The tower design sub-sequentially is changed in the following steps to support the grid plates. The new designs were also drafted in AutoCAD[®].

Design Step 1:

The width of both top and bottom grid plates was reduced, such that they were equal in size. The front surface of the grid plate was changed in order to couple the core with the crescent moderator as shown in Figure 4-3. Then, the tower design was changed to support the core from the grid plates. However, grid plates were only connected to the top and bottom grid plates from the back of the tower as shown in the figure. There were no bars to support the grid plates from the front side of the core. Therefore, it was necessary to install a support plate on top of the core.

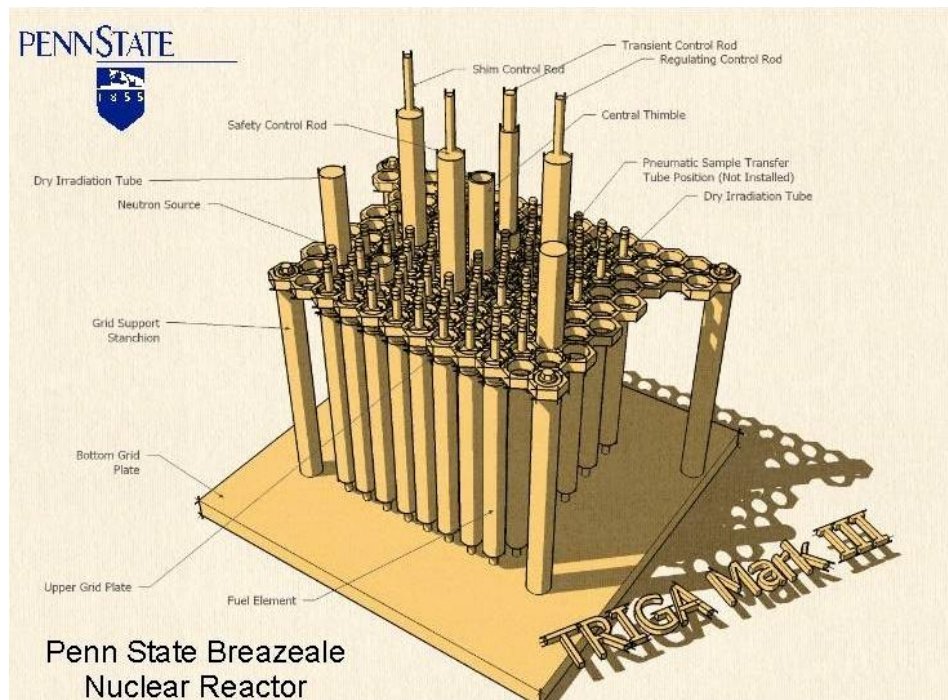


Figure 4-1: A 3D CAD drawing of the PSBR core, top and bottom grid plates, and fuel and control rods

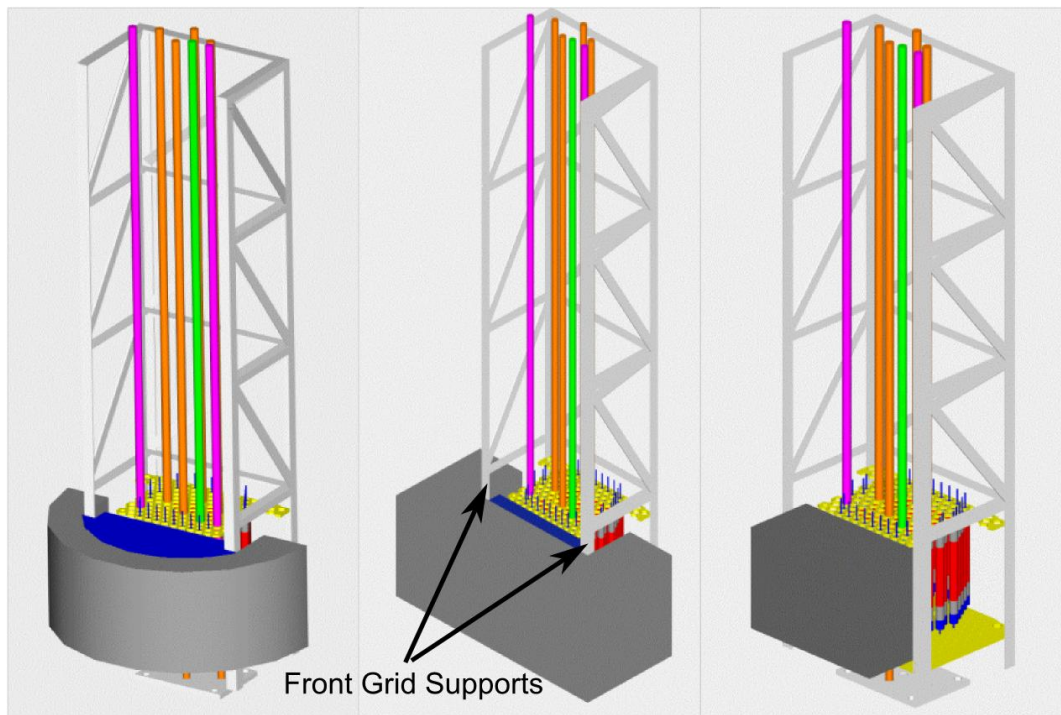


Figure 4-2: Suggested moderator shapes for the new moderator tank utilized with the existing reactor core design [Butler, reproduction]

Design Step 2:

Since the top and bottom grid plates were not completely supported by the suspension tower in the previous step, a support grid plate with holes was placed to the top of the core. New support bars were installed from the front side of the core (see Figure 4-4). Although, the core was mechanically established, the presence of a support grid plate would make the replacement of the fuel rods difficult, since each PSBR fuel rod is loaded from the top of the core. Additionally, the natural circulation of the water flow would be restricted by this plate. Therefore, placing a second grid plate on top of the core was dismissed.

Design Step 3:

The support grid plate introduced in Step 2 was changed by opening the top of the core in this step. Instead of a grid plate restricting the water flow and fuel transportation, two near support plates were installed as shown in Figure 4-5 to support the core from its front side. In this way, it was assured that reactor core would be fully supported by the tower.

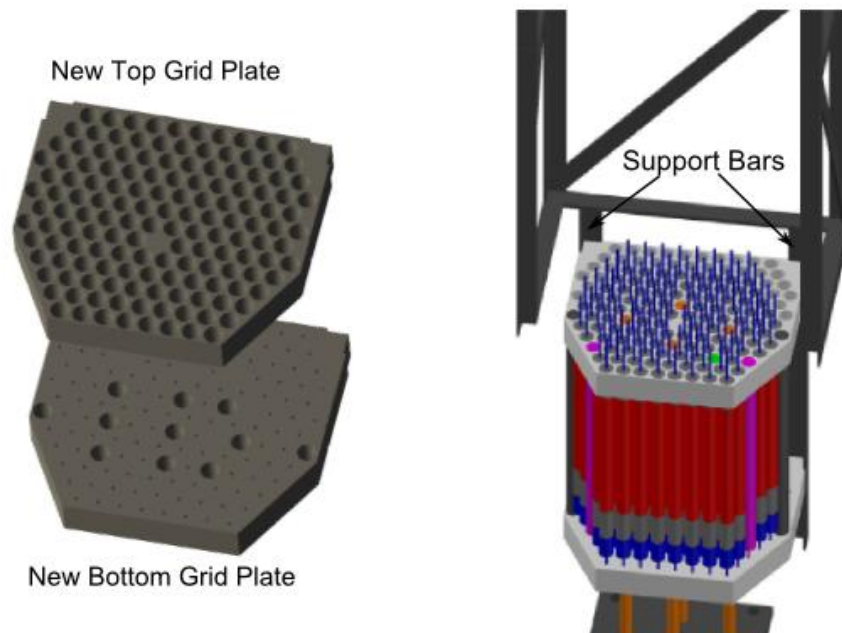


Figure 4-3: 3D AutoCAD® drawings of (left) new top and bottom grid plates for new PSBR core and (right) new support bars for the existing tower design

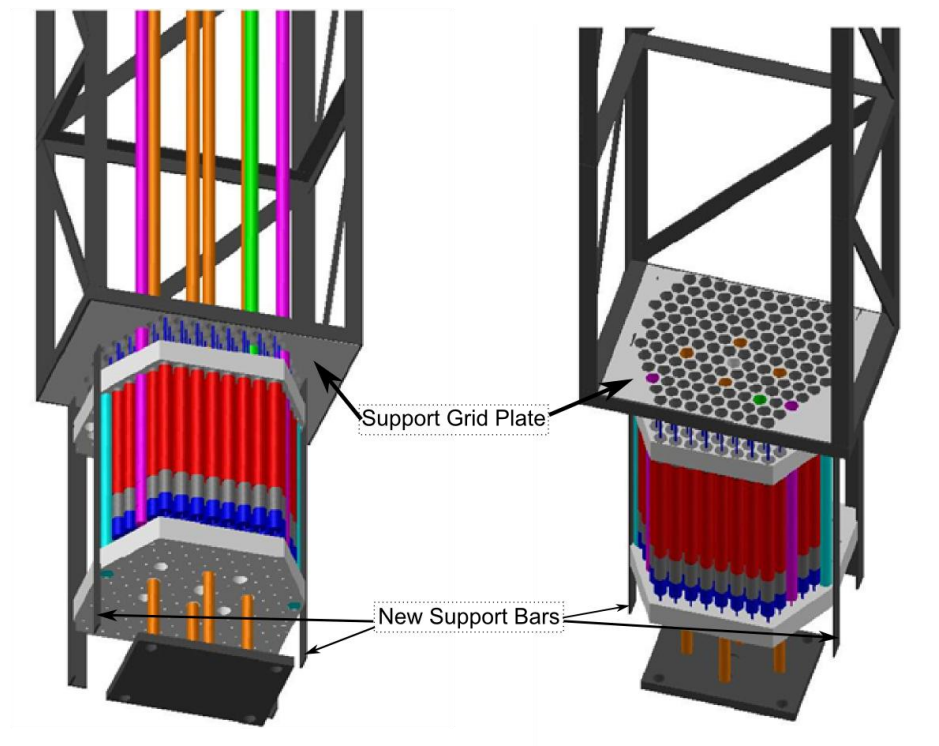


Figure 4-4: 3D AutoCAD[®] drawings of the new reactor core and existing tower designs with added new support grid plate and new support bars

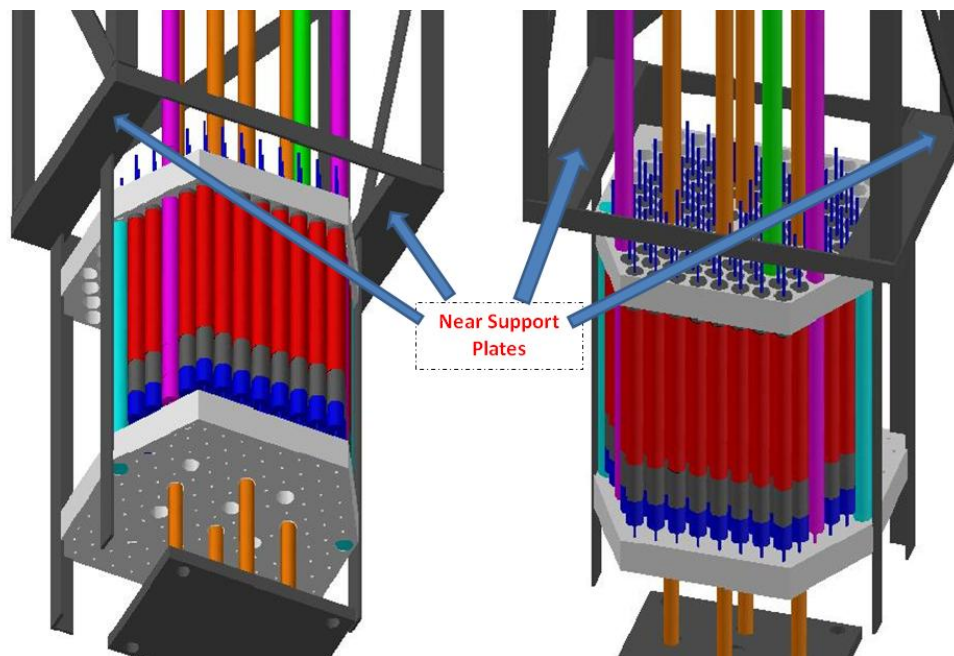


Figure 4-5: 3D AutoCAD[®] drawings of the new reactor core and existing tower with added near support plates

The next step in the design process was the coupling of the crescent moderator with the new core design. In order to achieve this, the shape of the moderator was modified at the core interface. The distance between the moderator and the core was set as 0.6 cm ($\frac{1}{4}$ inches) and the height of the moderator was set to 48.2 cm (19 inches) to cover the active length of the fuel elements. AutoCAD[®] drawings of the coupled assembly with the new tower design are given in Figure 4-6.

In the new configuration, the amount of pool water between the core and moderator interface was minimized. Hence, prompt gamma-ray generation by the hydrogen in water was reduced. A significant reduction in the number of hydrogen gamma-rays in the new beam port facilities is expected with this new configuration. The neutronic and thermal-hydraulics performance of the new design will be discussed in Chapters 6 and 7.

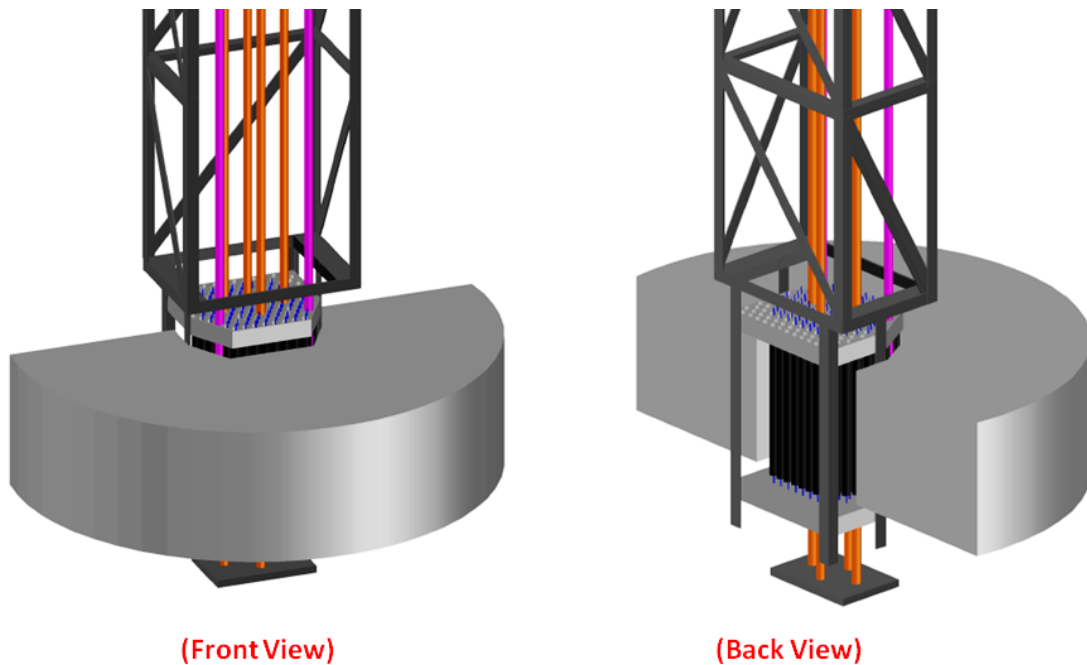


Figure 4-6: 3D AutoCAD[®] drawings of new core-moderator assembly

Design of New Beam Ports

There will be five new beam ports in the new design as mentioned in the previous section and one of them is going to be used for cold neutron source generation and transportation. Only the geometrical configuration of these beam ports with respect to core-moderator assembly was evaluated in this section. Detailed information about the design of each beam port will be given in Chapter 7. The proposed beam port configuration is shown in Figure 4-7. The relative orientation with the tilt angles of each beam port with respect to new BP2 (NBP2) was also calculated and given in this figure. The angles between the beam ports were calculated based on the assumption that the new core would be at the same location in the pool with the existing one. Neutron beam ports will be directed to the reactor core center and they will also be installed to the fuel element or core centerline.

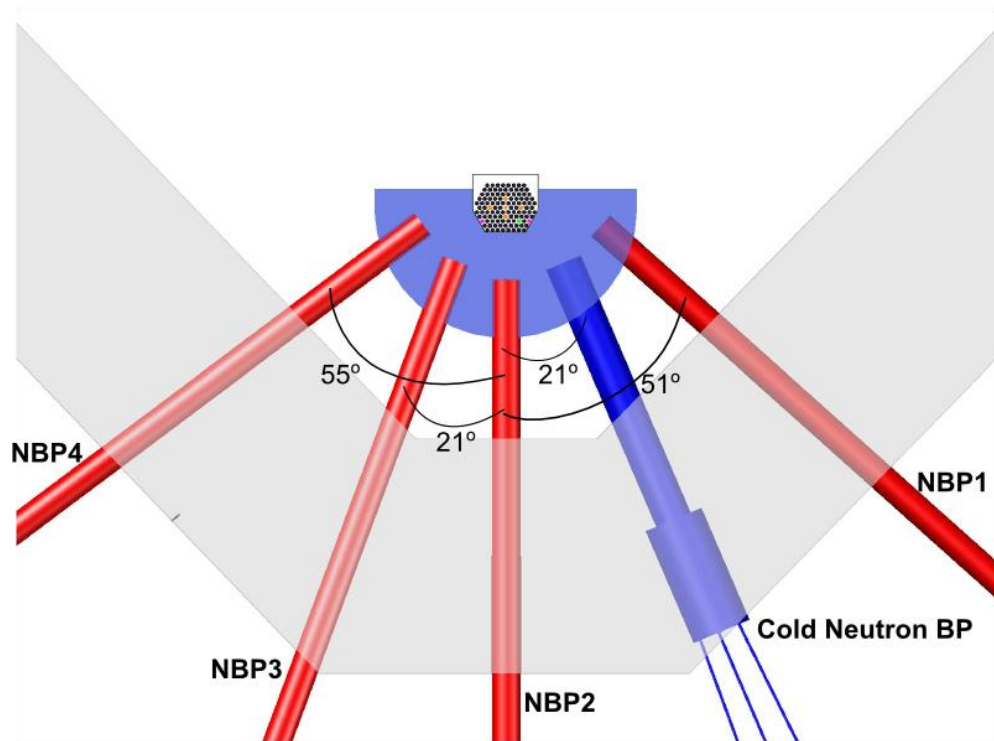


Figure 4-7: Proposed beam port layout with new core-moderator assembly design (Top View)

Chapter 5

NEUTRONIC ANALYSIS of NEW CORE-MODERATOR ASSEMBLY DESIGN of the PSBR

Neutronic analysis of the new PSBR core-moderator assembly was accomplished in three steps:

- i.** Determination of core excess reactivity.
- ii.** Optimization analysis.
- iii.** Calculation of neutronic performance of new core-moderator assembly.

First, excess reactivity of the core in the new PSBR design was calculated by MCNP5 simulation. Then, optimum moderator dimensions and beam port locations were determined by sensitivity studies. In the final part, the neutron and gamma flux spectra calculated by the MCNP simulations in the new beam port facilities were compared to the measurements in the existing facility to determine the neutronic performance of the new core-moderator assembly.

Core Excess Reactivity Analysis in the New Design

As mentioned in the previous chapter, the reference core loading was chosen as loading 53H at its initial load (May, 2009) in the MCNP calculations. However, graphite rods located at the periphery of the core will be removed in order to achieve full coupling between core and moderator. Thus, it is expected to see a decrease in core excess reactivity in new design by the reactivity contribution of graphite rods, which was measured as \$0.27. Since the existing D₂O tank and the graphite box will no longer be utilized in the new design, their reactivity contribution respectively measured as \$0.68 and \$0.03 will be deducted from the core excess reactivity in the new reactor. On the other hand, new crescent-shaped moderator tank is expected to increase the

core excess reactivity. In this section, excess reactivity of the core in the new reactor was analyzed by calculating the excess reactivity contribution of the new crescent-shaped moderator.

Two MCNP models were created for this purpose as shown in Figures 5-1, 5-2 and 5-3. The effective multiplication factor of the core in each model was calculated by making KCODE (criticality) calculations. The input parameters and assumptions made during these calculations are given as follows.

- i. Depleted fuel compositions in core loading-53H were used in the MCNP material definitions for the fuel and control rods models.
- ii. Control rods were modeled at all rods out (ARO) positions.
- iii. Fuel temperatures were set to 800 K in all fuel rods.
- iv. Criticality calculation was performed with 100 histories and 1 million particles per each history.

The depleted fuel compositions were obtained from the TRIGSIMS results which is the fuel management code system prepared by Tippayakul for the PSBR core [56]. Default ENDF/B-VII data tables were utilized during the MCNP calculations. Additionally, water temperature was set to 300K and S(α,β) treatment was applied for the hydrogen in light water, deuterium in heavy water (D₂O), and hydrogen and zirconium in ZrH in MCNP models. The MCNP calculations were independent from the control rod worth since ARO assumption was applied to both bare core model and core-moderator assembly model. Under these conditions, the estimated effective multiplication factors (k_{eff}) for each model are given in Table 5-1.

By using the MCNP predicted effective multiplication factor, the contribution of the new moderator shape to the core excess reactivity was determined using Equation (5-1).

$$excess\ reactivity\ (\$) = \frac{k_{eff1} - k_{eff2}}{k_{eff1}k_{eff2}\beta_{eff}}, \quad (5-1)$$

where k_{eff1} is the effective multiplication factor of the core, k_{eff2} is the effective multiplication factor of the core-moderator assembly and β_{eff} effective delayed neutron fraction, which is 0.007 for TRIGA reactors. By inserting calculated multiplication factors given in Table 5-1 into Equation (5-1), contribution of the new crescent moderator to the core excess reactivity was estimated as

$$excess\ reactivity\ (\$) = \$1.13 \quad (5-2)$$

with a standard deviation of 6×10^{-5} .

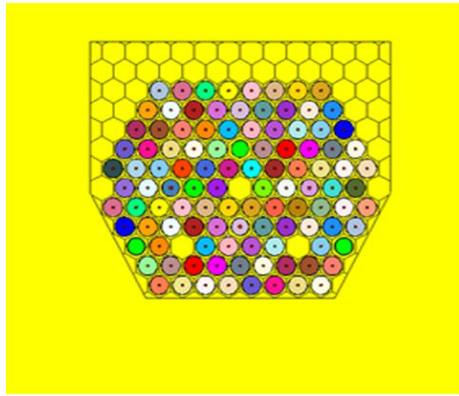


Figure 5-1: Core loading-53H in the new PSBR core model

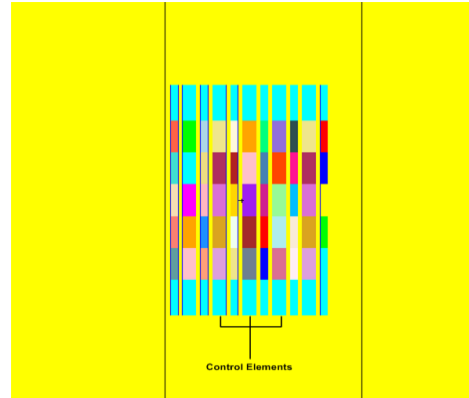


Figure 5-2: Fuel and control rods in the new PSBR core model

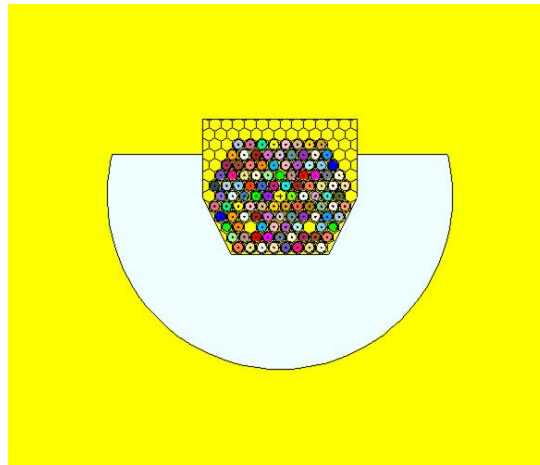


Figure 5-3: New PSBR core-moderator assembly model

By subtracting the decrease due to the removal of graphite elements, graphite box and the D₂O tank and by adding the increase due to the new moderator tank, a core excess reactivity gain in the new reactor was estimated as

$$\text{excess reactivity gain } (\$) = \$1.13 - \$0.27 - \$0.68 - \$0.03 = \$0.15 \quad (5-3)$$

with a standard deviation of 8.36×10^{-5} .

Table 5-1: MCNP calculated effective multiplication factors in bare core model and the core-moderator assembly model

Model	k_{eff}	Standard Deviation (σ)
Core	1.0345	3×10^{-5}
Core and moderator	1.0432	5×10^{-5}

Optimization Analysis

The optimization analysis in this study was a state-space search approach, analogous to exhaustive search [34, 35]. Exhaustive search is a widely used problem solving technique in which all possible candidates for the solution are defined and then each one is analyzed whether it will satisfy the problem's statement. Analogous to this approach, state is defined as each possible configuration in the search space defined as all possible states. The objective of this approach is to reach to an optimum state in a defined search space.

In the optimization study, there are some constant parameters which cannot be altered during the state-space search. These parameters are listed as follows.

- i. The alignment of new beam ports with respect to core centerline

This is mainly limited by the available floor space in the beam hall. As mentioned in the previous chapter, each beam port will be utilized for a specific experimental method and

corresponding instrument in the beam hall. It is not possible to change the location of these facilities due to space constraints in the beam hall. Besides that, the new core will be at its current location in the pool. Therefore, the alignments of the new beam ports in the new core-moderator assembly are not changed during the study. The geometrical configuration of the new beam ports with the tilt angles with respect to NBP2 are previously given in Chapter 4 and will be utilized in the new reactor.

ii. Location of the new beam ports with respect to pool floor

The axial power-density profile of fuel elements is a cosine distribution about the center of active fuel length. Thus, the relative location of all the beam ports with respect to pool floor will be at the center of the fuel elements to obtain the maximum neutron flux in the beam port facilities.

iii. The height of the new moderator tank

The height of the new moderator tank was set to 48 cm (19 inches) to increase the cross-flow into the core. In contrast to the existing moderator tank, the new moderator tank will cover more than half of the core's circumference. It is expected to observe a significant decrease in the cross-flow. In order to provide the necessary cooling in the core, it was decided to provide 5 cm (2 inches) openings between the top and bottom grid plates and the new moderator tank to allow the passage of pool water into the core.

By taking into account these constant parameters, two possible states were identified as radius of the moderator and distance between each new beam port and the core face in the optimization study. These two parameters are related to the moderation area in the new moderation tank. Control parameters during the optimization were selected as thermal-to-fast neutron ratio and the gamma dose. The main purpose of the optimization study was to calculate the optimum design parameters which will maximize the thermal-to-fast neutron ratio while mitigating the number of gamma rays in the new beam port facilities.

Calculation Methodology

MURE code was utilized in the optimization study [57]. MCNP inputs and models are automatically generated by MURE code depending on the user's inputs. The advantage of using MURE code in this study is the ability to make easy design changes in the optimization or the search parameters while keeping all the other design parameters same.

The input parameters and assumptions used in the MCNP calculations are listed below.

- i.** Fresh fuel assumption
- ii.** Control rods were in ARO state
- iii.** Criticality calculation with 5000 histories and 1 million particles per history.
- iv.** The temperature of the fuel elements and pool water were set to 800 K and 300 K, respectively.
- v.** Two steps calculations:
 - a.** Surface source fabrication in “full core” model using “conical source biasing” technique
 - b.** Fixed source calculation in “beam port” model using the fabricated surface source and point detector estimators

Fresh fuel assumption was preferred during the optimization analysis because the optimization/search parameters were calculated in a comparative case study. For this purpose, all the other parameters such as core loading pattern, fuel and control rod models, core-moderator configuration in the pool, etc. were kept constant during the MCNP calculations. This allows the determination of relative increase or decrease in the thermal-to-fast neutron ratio and gamma dose (control parameters) in the new neutron beam ports as the search space is canvassed. Also the fresh fuel assumption significantly shortened the computational time and allowed the analysis of more cases in MCNP simulations.

However, the distance between the core and the tally locations at the end of each new beam port would not allow obtaining good statistics in the tally results. For example, the distance between the core and tally location in new BP2 (NBP2) is approximately 3 meters. The fission neutrons escape from the core in any direction. However, the neutrons entering into each new beam port, which is a very small fraction of the total numbers of fission neutrons, do not penetrate to the end of the new beam ports with reasonable weights and accuracy. As mentioned in Chapter 1, non-analog Monte Carlo methods or variance reduction techniques are utilized in MCNP calculations for the solution of source-tally problems, also known as “deep penetration” problem [12].

In this study, two special variance reduction techniques, named “conical source direction biasing” and “point detector” estimator, were utilized to overcome the deep penetration problem. The calculation methodology can be summarized as follows. The neutron and gamma spectra at the end of beam ports were calculated in two steps. First, a criticality calculation was performed in the “full core” model to fabricate a source on a hypothetical surface defined at a depth of ~1.2 meters of each new beam port model. Angular and spatial distributions of neutrons and gamma-rays passing through this surface were tallied in this model. Second, a separate fixed source calculation was performed in each individual “beam port” model starting from the hypothetical surface source. In the conical source direction biasing technique, source emission is restricted to a set of nested cones about the bias or reference direction as shown in Figure 5-4. Any specified cone divides the angular domain into two pieces: inside of cone and outside of cone. All the particles inside of the cone have identical weights and all the particles outside of the cone have identical weights. The fraction of particles started in any cone is specified by source probability (SP) and source bias (SB) cards in the SDEF (general source) card in MCNP input. In this study, conical source direction biasing parameters were calculated in the criticality analysis. Then, the neutron and gamma flux spectra was calculated by defining point detectors at the end of the beam

port models in existing and new PSBR designs. As an example, the angular and spatial neutron current spectra obtained on the source surface in NBP2 are shown in Figures 5-5 and 5-6, respectively. The angular neutron current spectrum shows that almost all of the neutrons are moving in the forward direction towards the end of the beam port on the surface source. In addition the particles passing through the tally surface in NBP2 are almost spatially uniform. Similar angular and spatial neutron current spectra were observed in the other proposed new beam ports.

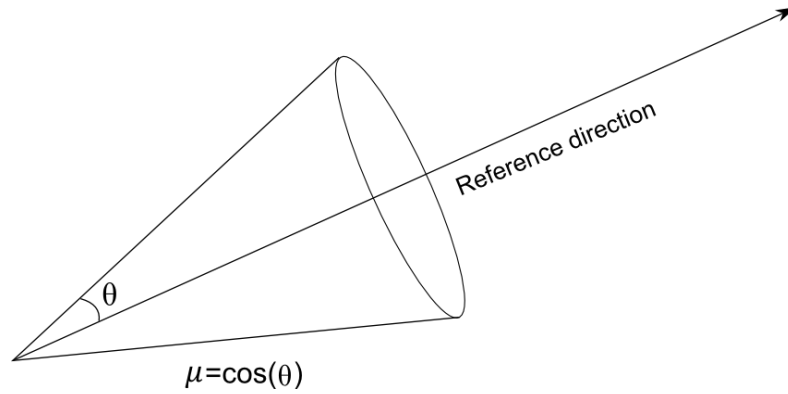


Figure 5-4: Source bias in a cone within a reference direction

The point detector estimator in MCNP is mainly utilized to transport particles to a tally region where it is very difficult or impossible to transport the particles in MCNP calculations. It is a deterministic estimate of particle flux at a point in space. During the random walk of source particles, contribution of each particle directly to the point detector is estimated at each collision or source event and a “pseudoparticle” is scored at the point detector with a weight of w given as

$$w = \frac{w_o p(\mu) e^{-\lambda}}{2\pi R^2}. \quad (5-4)$$

In this equation, w_o is the exit weight of the source particle, $p(\mu)$ is the value of the point detector function after the scattering directly toward the point detector, λ is the mean free path through all regions between the collision and the point detector, and R is the distance between the location where collision takes place and the point detector. The point detector estimator is computationally

very expensive because the contribution of each pseudoparticle to the point detector must be followed during its trajectory. Nevertheless, it enormously increases the number of particles reaching to the tally region.

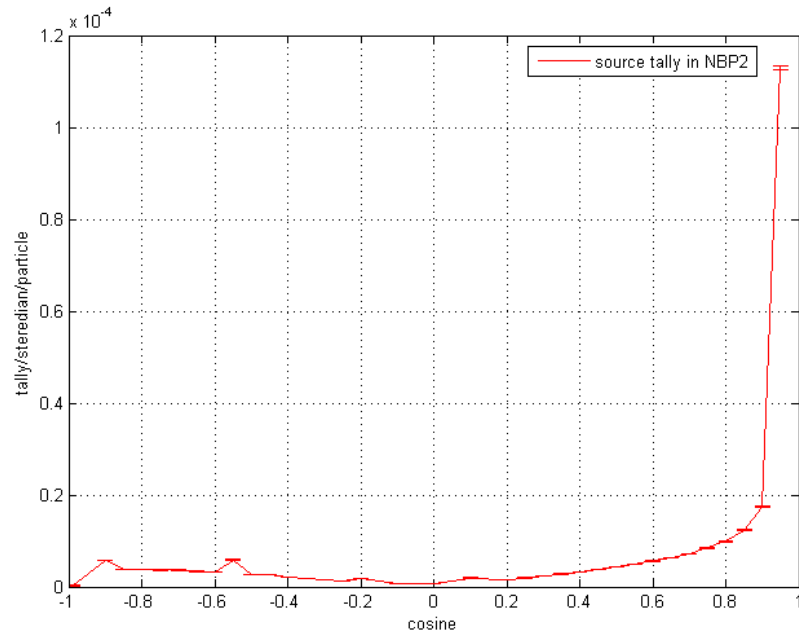


Figure 5-5: Angular neutron current spectrum obtained on source surface in NBP2

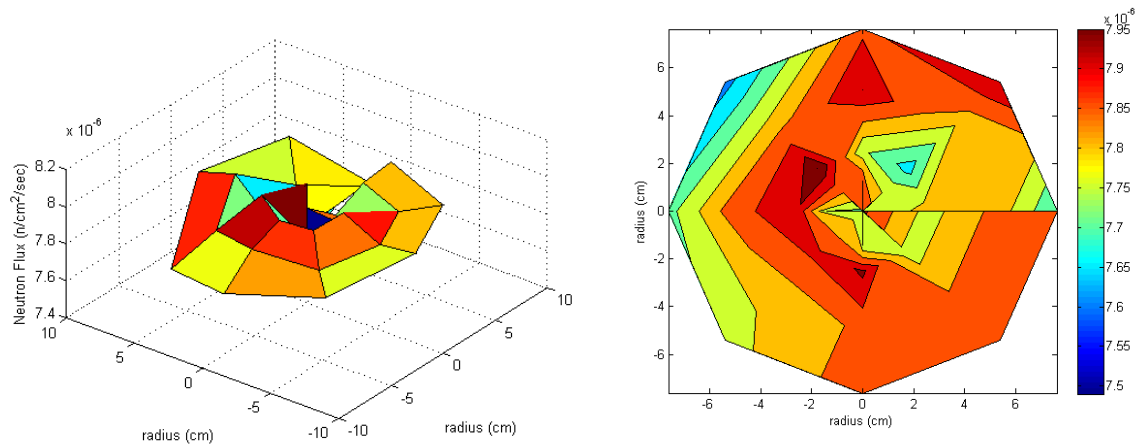


Figure 5-6: Spatial neutron current spectrum obtained on source surface in NBP2

The accuracy of point detector estimator and conical source direction biasing technique was first verified in the existing PSBR design models previously given in Figures 3-8 and 3-9. The neutron flux spectrum at the existing BP4 exit was calculated with the input parameters and assumptions given before. Then MCNP predicted total thermal neutron flux (<1 eV) was compared with the measurements obtained with a single-disk-slow-chopper time-of-flight (TOF) spectrometer by Niederhaus [58]. This method is preferred instead of comparing with thermal neutron flux measurements with activated gold foils, seen in Chapter 3, because the definition of thermal neutron flux measured with this technique is different than the integral form of the thermal neutron flux calculated by MCNP code as discussed in Section 3.4.

MCNP input files created by the TRIGSIMS code during the burnup calculations was used. Therefore, input files had the entire necessary information such as depleted fuel element, control rod compositions, etc. to accurately calculate the neutron and gamma flux spectra at the beam port facilities. Beside that the materials in the PSBR are already defined with proper extensions in MAT cards in MCNP input. These extensions are associated with the temperature dependent cross section data set generated for the PSBR core. In the input files, the fuel temperatures in all the fuel and control rods and the water temperature in the core are set to 800 K and 300 K, respectively.

The flux tallies are given in normalized fraction per starting source particle in MCNP output. A proper scaling factor is required to convert them to actual results. The scaling factor for the neutron flux can be calculated using Equation 5-5.

$$S = \frac{P(W)\vartheta \left[\frac{\text{neutron}}{\text{fission}} \right]}{\left[1.602 \times 10^{-13} \frac{\text{J}}{\text{MeV}} \right] \omega_f \left[\frac{\text{MeV}}{\text{fission}} \right]} \times \frac{1}{k_{eff}}, \quad (5-5)$$

where $P(W)$ is the operating power, ϑ is the number of neutrons per fission, ω_f is the energy released per fission and k_{eff} is the effective multiplication factor of the core. For the PSBR in a 1-MW steady-state power operation, the scaling factor was determined as

$$S_N = \frac{1 \times 10^6 (W) 2.439 \left[\frac{\text{neutron}}{\text{fission}} \right]}{\left[1.602 \times 10^{-13} \frac{J}{MeV} \right] 200 \left[\frac{MeV}{\text{fission}} \right]} \times \frac{1}{1.00} = 7.62 \times 10^{16} \text{ neutrons/s} \quad (5-6)$$

The scaling factor for the gamma flux was predicted by using the measured gamma-ray intensity at the BP4 exit (location #3) in the existing reactor. The gamma-dose in the neutron beam laboratory was measured as 24.3 R/h (see Table 3-6). These measurements were conducted by using Ludlum gamma monitor for 1 MWt reactor power. During the gamma-dose measurements, no collimator was present in the BP4. In order to convert the dose unit from mR/hr to counts per second (cps), manufacture specified conversion factor, 18 cpm/(mR/hr) was employed. The number of gamma-rays in the beam laboratory was calculated as

$$J_\gamma = 24300 \text{ mR/hr} * 18 \frac{\text{cpm}}{\text{mR/hr}} * \frac{1 \text{ min}}{60 \text{ s}} = 7290 \text{ counts/s} \quad (5-7)$$

The gamma-dose was measured as 8.2 R/h at the end of BP4 (location #3) when the collimator was in BP4. After the unit conversion to cps, the total gamma intensity at the collimator exit was calculated as

$$J_\gamma = 8200 \frac{\text{mR}}{\text{hr}} * 18 \frac{\text{cpm}}{\frac{\text{mR}}{\text{hr}}} * \frac{1 \text{ min}}{60 \text{ s}} = 2460 \frac{\gamma}{\text{s}} \quad (5-8)$$

The error in the measurements was assumed to be 10% in accordance with the manufacturers' specifications for the Ludlum gamma monitor. By using the MCNP predicted normalized total gamma flux at location #3, the measured total gamma intensity, and the beam diameter at the collimator exit; the scaling factor for gamma flux was calculated using the following equation.

$$S_{\gamma} = \frac{J_{\gamma}}{\Phi_{\gamma} * A}. \quad (5-9)$$

The total gamma flux in normalized form at location #3 was estimated by the MCNP simulation as

$$\Phi_{\gamma} = 9.19 \times 10^{-11} \pm 1.05 \times 10^{-12} \frac{\gamma}{cm^2 - sec}. \quad (5-10)$$

Then, the scaling factor for the gamma flux at 1 MWt steady-state power was estimated as

$$S_{\gamma} = 8.52 \times 10^{12} \pm 8.57 \times 10^{11} \gamma/s \quad (5-11)$$

By using the scaling factors given in Equation (5-6) for neutrons and in Equation (5-11) for gamma-rays, the total thermal neutron flux ($E < 1$ eV) at the end of existing BP4 was calculated by the MCNP simulation as

$$\Phi_{thermal} = 2.86 \times 10^7 \pm 3.45 \times 10^5 \frac{n}{cm^2 - sec}. \quad (5-12)$$

The total fast neutron flux ($E > 0.1$ MeV) at the BP4 exit was calculated as

$$\Phi_{fast} = 2.76 \times 10^6 \pm 4.02 \times 10^5 \frac{n}{cm^2 - sec}. \quad (5-13)$$

The total gamma flux at the same tally location was determined as

$$\Phi_{\gamma} = 782.98 \pm 79.29 \frac{\gamma}{cm^2 - sec}. \quad (5-14)$$

The MCNP predicted total thermal neutron flux is within the acceptance range of measurements with slow-chopper time-of-flight spectrometer, which is $3.0 \times 10^7 \mp 0.3 \times 10^7 \frac{n}{cm^2.s}$. This verifies the accuracy of the variance reduction techniques employed in the study. In addition to total thermal neutron flux, the neutron flux spectrum at the end of BP4 is given in Figure 5-7. The observed difference between the measured and calculated results is caused by the fact that the fuel temperatures in the calculation were set to 800 K in the MCNP inputs. The TRIGA fuel is affected by the temperature increase in the fuel such that up-scattering probability by the hydrogen in U-ZrH fuel matrix increases with the temperature rise. This significantly changes the

neutron economy in the core. Therefore, the correct definitions of fuel temperatures in MCNP models can improve the simulation results.

Additionally the gamma flux spectrum calculated at the end of existing BP4 at 1-MWt power is given in Figure 5-8. The 2.2-MeV hydrogen gamma peak was barely observed in this spectrum. After the verification of the results, this calculation approach was followed in the MCNP calculations throughout the thesis.

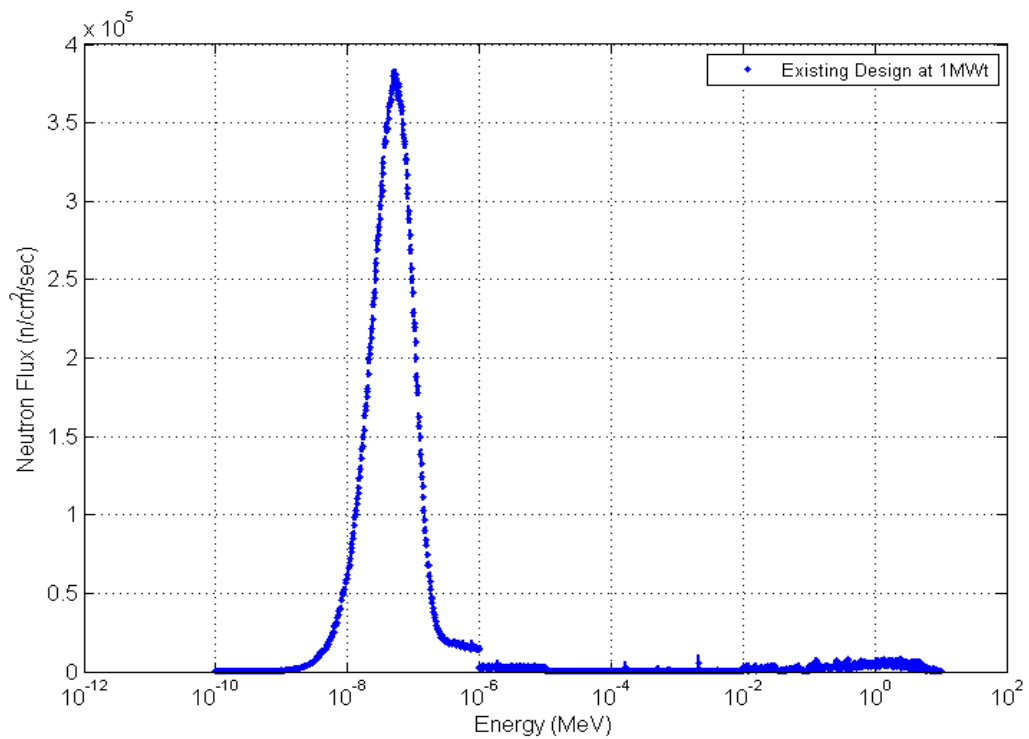


Figure 5-7: Neutron flux spectrum calculated at BP4 exit in the existing PSBR design at 1 MW

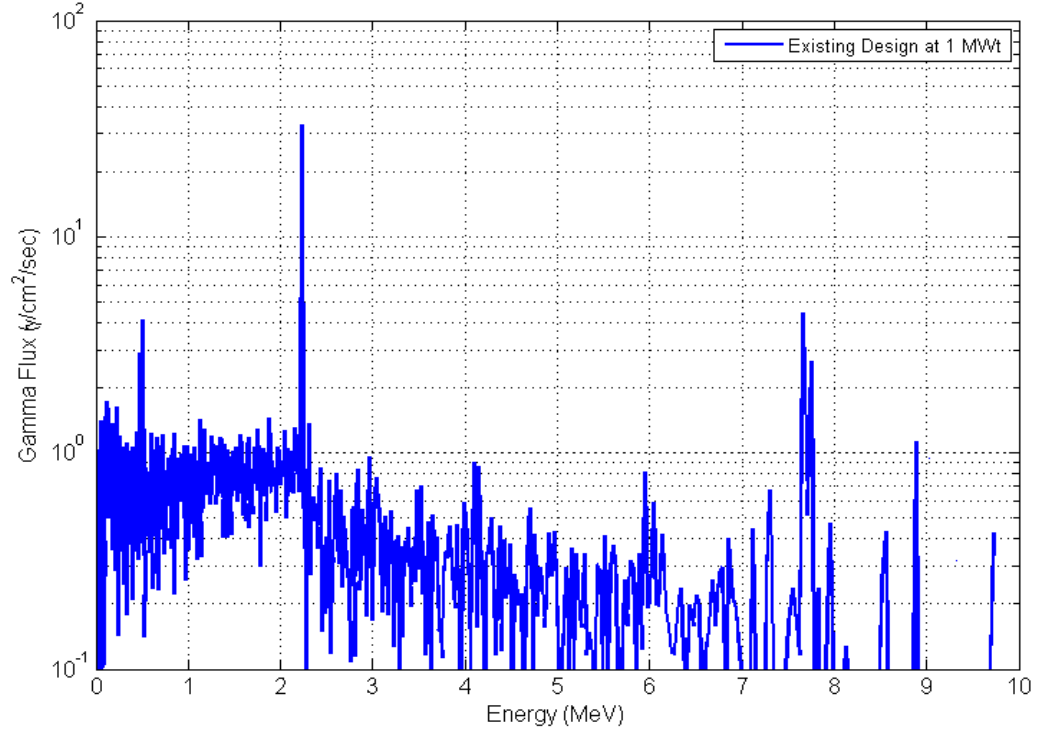


Figure 5-8: MCNP predicted gamma flux spectrum at BP4 exit in the existing PSBR design at 1MW

Calculation of Optimum Beam Port Locations

As mentioned in the previous section, there are two constant parameters which restrict the calculation of optimum beam port locations in the new PSBR design. The first parameter is the relative orientation of new beam ports with respect to the reactor core. The second one is the relative axial location of beam ports along the active length of fuel elements. It is known that axial fuel power profile follows the cosine distribution about the fuel element centerline [28]. That is why the proposed new beam ports will be installed at the core centerline or at the midplane of the fuel rods. Thereafter, the only parameter that can be altered is the distance between the beam ports and the core. This distance is important as the thermalization area in the moderator will change depending of this distance for each beam port. One of the targets in this

study is set out to find the optimum beam port to core distance that will provide maximum thermalization to the fast neutron beam in the moderator. In this analysis, the effect of the distance change on the neutronic performances of the new beam ports was evaluated in successive MCNP calculations.

In the MCNP calculations, fresh fuel and all control rods out assumptions was employed to shorten the computational time. The fuel temperatures in all the fuel and control rods and water temperature in the pool were set to 800 K and 300 K, respectively. For the sake of simplicity in the calculations, the new beam ports were modeled as empty tubes with 15.24-cm (6-inches) inner diameters. As a first guess, the moderator radius was set to 76.2 cm (30 inches).

MCNP simulations were performed in two-stages with conical source direction biasing and point detector estimator variance reduction techniques. Since there is a two-fold symmetry in the core, the neutron and gamma flux spectra were calculated in only New BP2 (NBP2), New BP3 (NBP3) and New BP4 (NBP4). It was assumed that simulation results for New BP1 (NBP1) and cold neutron BP will be identical to NBP4 and NBP3, respectively. The MCNP models used in these calculations are shown in Figure 5-9. The arrows in these figures indicate the direction of position and hence the distance change. Three hundred histories with 1-million particles per history were sampled in the criticality calculations of the full core model. A total of two hundred million neutrons were sampled on the fixed source in the beam port models. The MCNP predicted relative error was less than 5% in the tally results which is the acceptable error for the point detector tally.

The total thermal (<1 eV) and fast (>0.1 MeV) neutron flux calculated in NBP2, NBP3 and NBP4 as a function of distance are shown in Figures 5-10, 5-11 and 5-12, respectively. The total thermal and total fast neutron flux estimated at the end of new beam ports were given individually instead of the thermal-to-fast neutron ratio in these figures. This was necessary

because a steady increase was observed in the thermal-to-fast neutron ratio although the total neutron flux was decreasing after a certain distance.

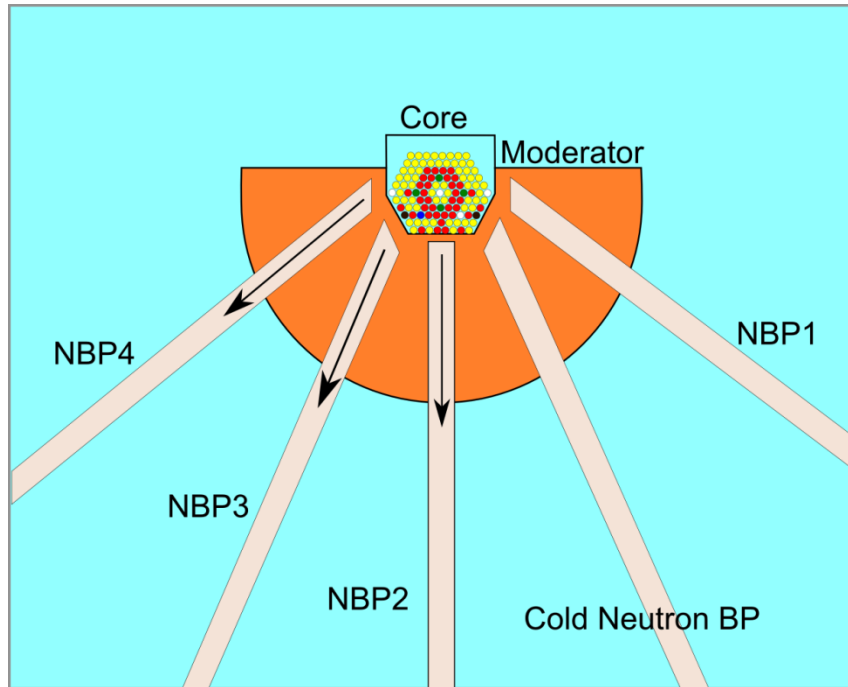


Figure 5-9: MCNP models employed in the optimum distance calculations for NBP2, NBP3 and NBP4

In these figures, new beam ports retreat starting from the core face which corresponds to zero in the x-axis by increasing the thermalization area in the moderator tank. Therefore, the total fast neutron flux at the tally locations decreases as expected because of moderation. As more fast neutrons become thermal, number of thermal neutrons entering into the beam port increases. However, this will not be a constant increase. After a specific distance, a fraction of thermal neutrons will be scattered out from the beam port direction in the moderator tank by interacting with the deuterium. This fraction will be higher than the fast neutron thermalization rate as the distance between the beam port and the core face increases. Therefore, the thermal neutron flux in each new beam port model will decrease after a specific distance, which is specific to the geometrical configuration of the beam port with respect to new core-moderator assembly.

By using the same calculation methodology, the total gamma flux as a function of distance change was estimated in new beam ports. The results are shown in Figures 5-13, 5-14 and 5-15. MCNP analyses results clearly show that as the distance between the core faces and new beam ports increases, the thermal-to-fast neutron ratio increases and the gamma flux decreases in the new beam ports. On the other hand, the total thermal neutron flux becomes maximal at 8 cm for NBP2, 12 cm for NBP3, after 4 cm for NBP4. In order to utilize the highest thermal neutron beam to the experimental facilities, the optimum distance between core and each new beam port were selected as tabulated in the Table 5-2. At these selected distances, total thermal neutron flux will be slightly less than the maximum value and the total fast neutron and total gamma flux will be reduced significantly in the new beam ports.

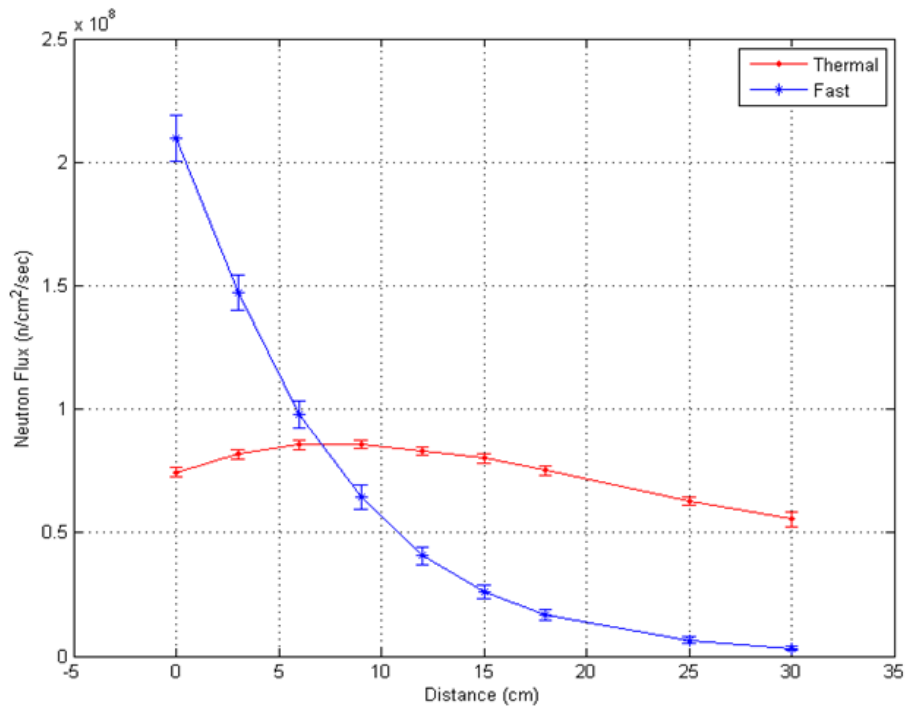


Figure 5-10: Total thermal and fast neutron flux as a function of distance between the core face and NBP2

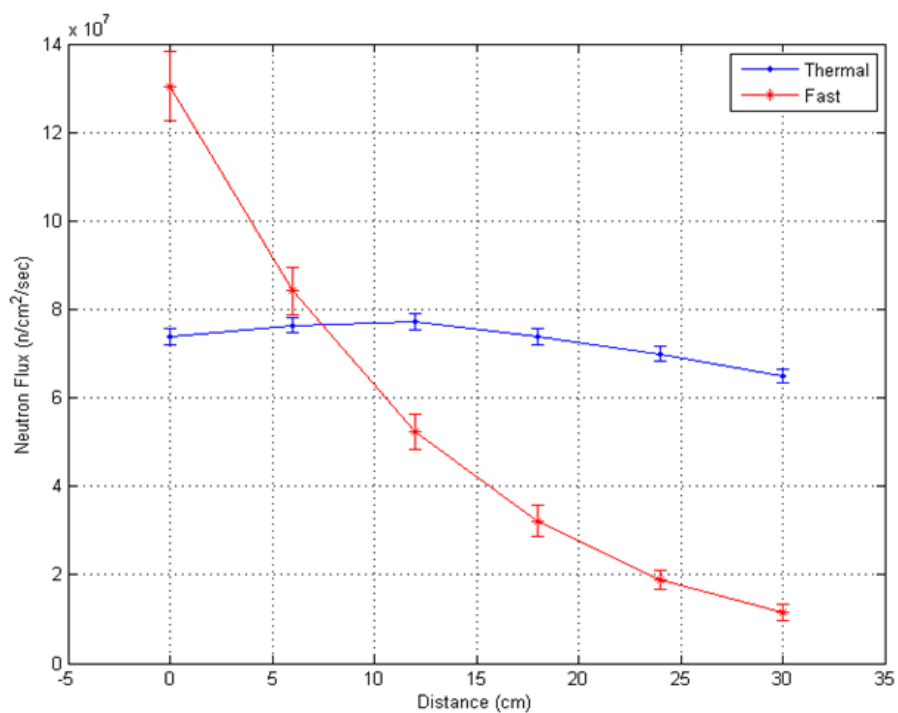


Figure 5-11: Total thermal and total fast neutron flux as a function of distance between the core face and NBP3

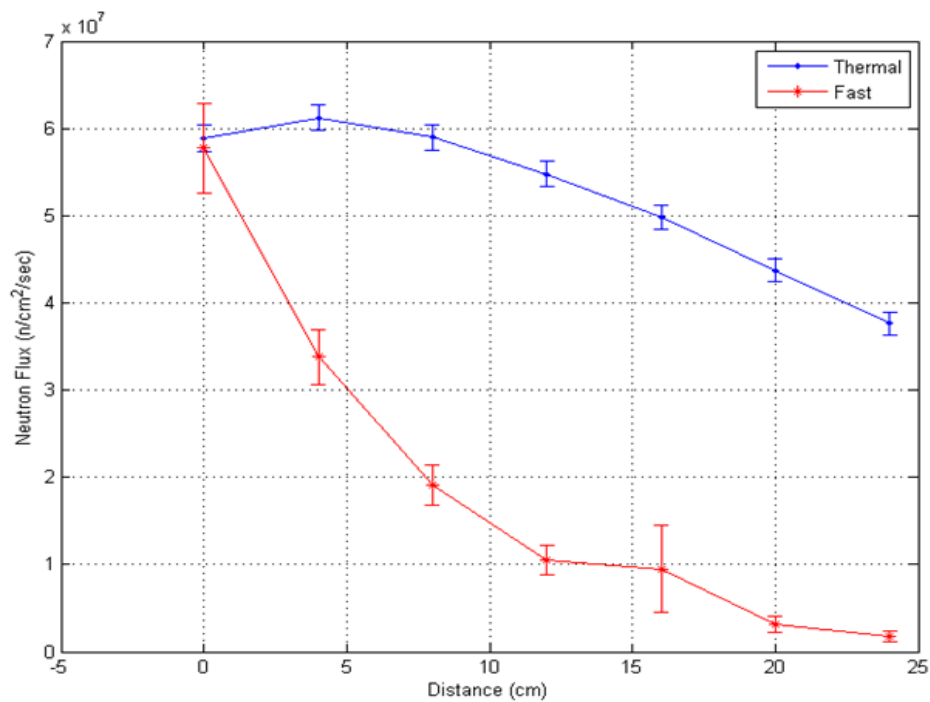


Figure 5-12: Total thermal and total fast neutron flux as a function of distance between the core face and NBP4

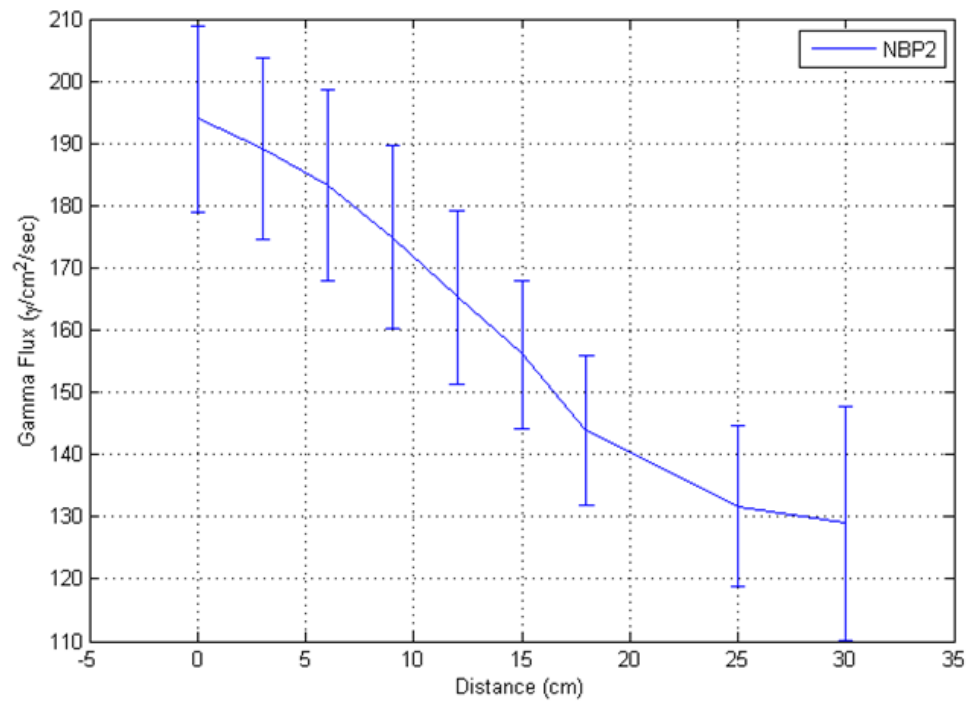


Figure 5-13: Total gamma flux as a function of distance between the core face and NBP2

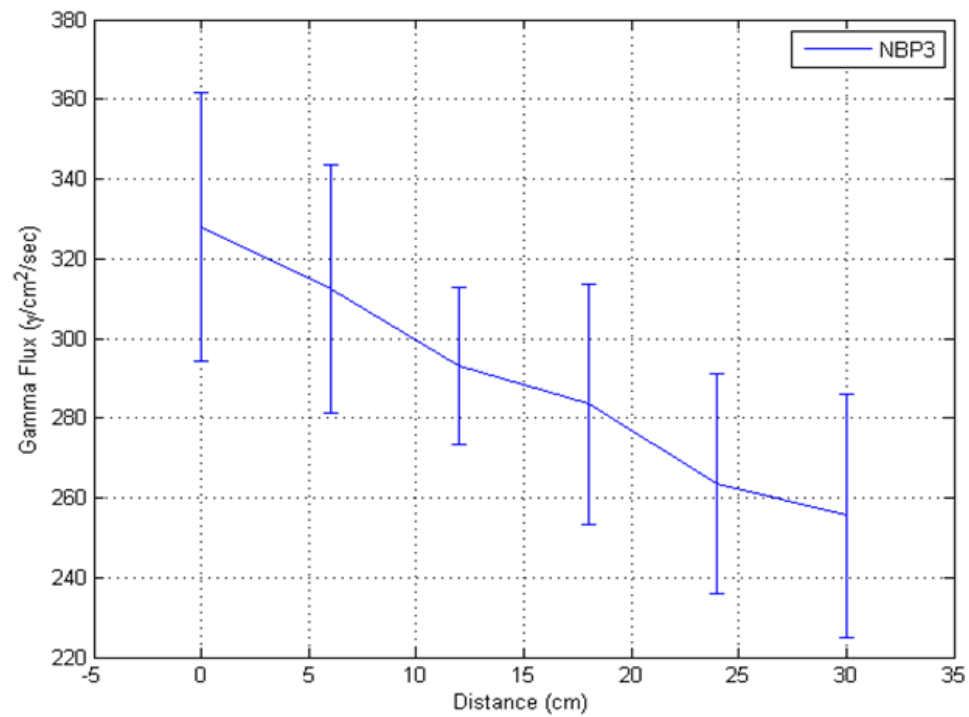


Figure 5-14: Total gamma flux as a function of distance between the core face and NBP3

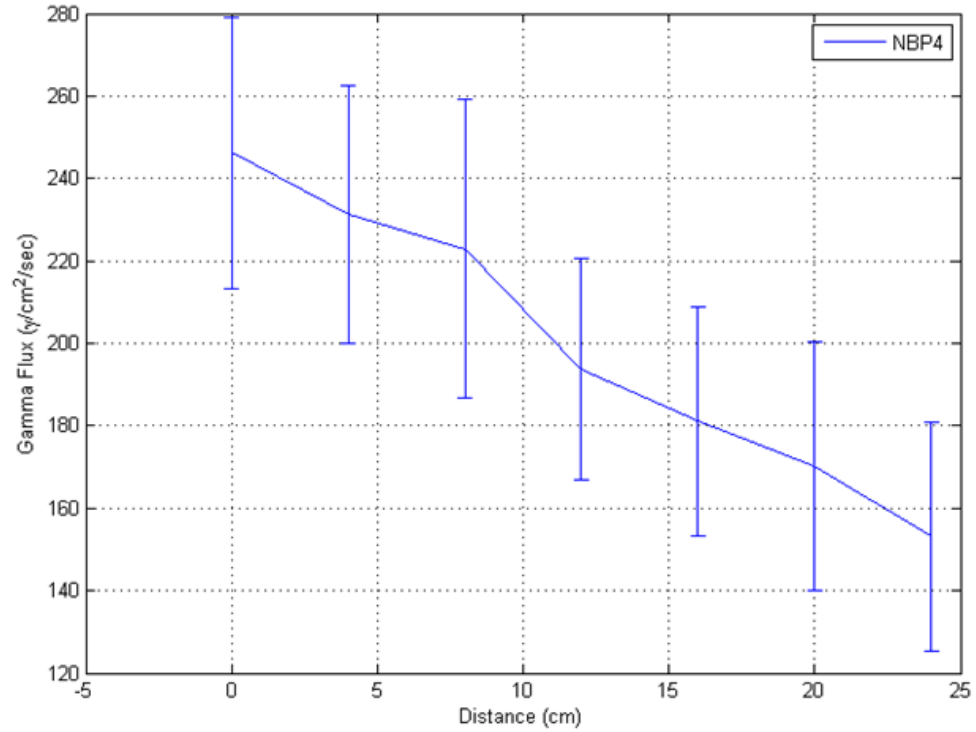


Figure 5-15: Total gamma flux as a function of distance between the core face and NBP4

Table 5-2: MCNP predicted optimum distance values between the core faces and the new beam ports

New Beam Port	Optimum Distance (cm)
#1	12
Cold neutron source	18
#2	15
#3	18
#4	12

Calculation of Optimum Moderator Dimensions

The purpose of the optimization study for the new crescent-shaped moderator tank was to determine its optimum size in radial direction (radius) since its length had been set to 48.2 cm (19

inches) in order to maximize the cross-flow entering into the core. The center of the moderator tank is the location of the centre thimble in the core. Similar to the previous study, MURE code was employed to generate MCNP inputs. Then, the effect of size change in moderator tank to the neutron and gamma flux spectra in the new beam port facilities was evaluated through MCNP simulations. Calculation methodology for this case is summarized as follows. By changing the radius of the moderator tank in successive MCNP calculations, the neutron and gamma flux spectra were calculated at the end of each new beam port model. All the other model parameters were kept constant. Figure 5-16 shows the size change in the models used this analysis. Similar to the previous cases, MCNP simulations performed in two-steps with full core and individual beam port models were followed with aforementioned two variance reduction techniques. Three hundred active cycles with 1-million particles per cycle were sampled in the criticality calculations. A total of two hundred million neutrons were sampled on the fixed source calculation. Control rods were modeled with all rods out (ARO) assumption and fuel and water temperatures were set to 800K and 300K, respectively.

The beam ports were modeled at their optimum distances calculated in the previous section. The results of the MCNP calculations to estimate the optimum moderator size are given in Tables 5-3, 5-4 and 5-5 for the total thermal ($<1\text{eV}$) and total fast flux ($>0.1\text{MeV}$) at different moderator sizes.

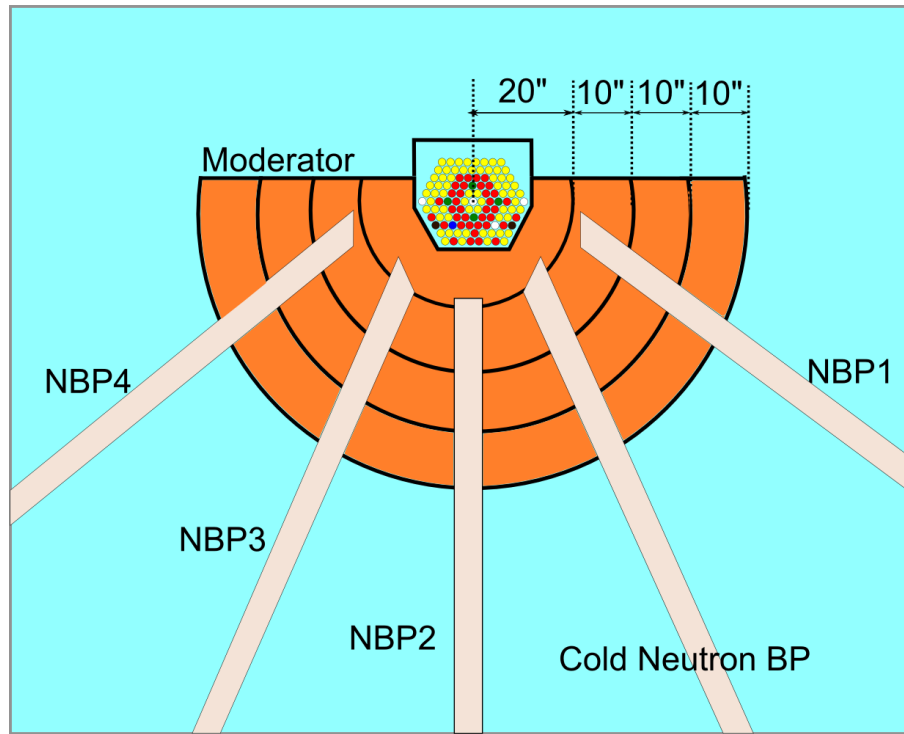


Figure 5-16: MCNP models created in the optimum moderator size calculations

Table 5-3: The total thermal neutron flux calculated at the end of NBP2 for different moderator sizes and 1MWt power

Moderator Size (cm)	Total Thermal Neutron Flux (10^7 n/cm ² /sec)	Error (10^6 n/cm ² /sec)	Total Fast Neutron Flux (10^7 n/cm ² /sec)	Error (10^6 n/cm ² /sec)
25.40	4.50	±1.34	2.09	±2.29
38.20	6.25	±1.46	2.48	±2.26
50.80	7.28	±1.81	2.55	±2.50
63.50	8.00	±1.95	2.58	±2.52
76.20	8.82	±2.11	2.62	±2.52
88.90	8.94	±2.06	2.66	±2.48
101.60	8.99	±2.36	2.66	±2.46
114.30	9.01	±2.36	2.66	±2.46

Table 5-4: The total thermal neutron flux calculated at the end of NBP3 for different moderator sizes and 1MWt power

Moderator Size (cm)	Total Thermal Neutron Flux (10^7 n/cm²/sec)	Error (10^6 n/cm²/sec)	Total Fast Neutron Flux (10^7 n/cm²/sec)	Error (10^6 n/cm²/sec)
25.40	1.07	±0.75	1.44	±3.41
38.20	5.24	±4.30	2.98	±5.49
50.80	6.31	±4.30	3.14	±5.48
63.50	7.38	±1.82	3.21	±5.63
76.20	8.53	±2.10	3.24	±5.68
88.90	9.71	±2.36	3.25	±5.60
101.60	10.30	±2.41	3.35	±5.89
127.00	10.90	±2.57	3.32	±5.89
152.40	10.90	±2.57	3.32	±5.89

Table 5-5: The total thermal neutron flux calculated at the end of NBP4 for different moderator sizes at 1MWt power

Moderator Size (cm)	Thermal Flux (10^7 n/cm²/sec)	Error (10^6 n/cm²/sec)	Fast Flux (10^7 n/cm²/sec)	Error (10^6 n/cm²/sec)
25.40	0.96	0.56	0.98	2.36
38.20	4.12	1.26	1.77	3.56
50.80	5.89	1.57	1.91	3.78
63.50	6.60	1.73	1.93	3.93
76.20	7.54	2.01	1.94	3.98
88.90	8.19	2.49	1.98	4.06
101.60	8.44	2.68	1.99	4.15
139.70	8.69	2.68	1.96	4.01
165.10	9.06	4.03	1.99	4.08

The calculated thermal-to-fast neutron ratio as a function different moderator size are given in Figures 5-17, 5-18 and 5-19 for each new beam port model.

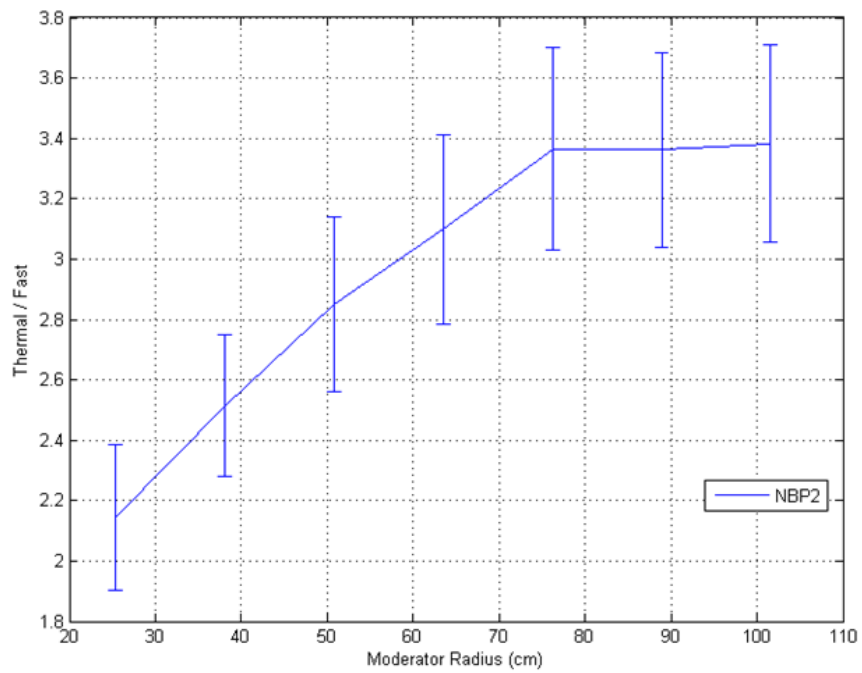


Figure 5-17: Thermal-to-fast neutron ratio as a function of moderator size in NBP2

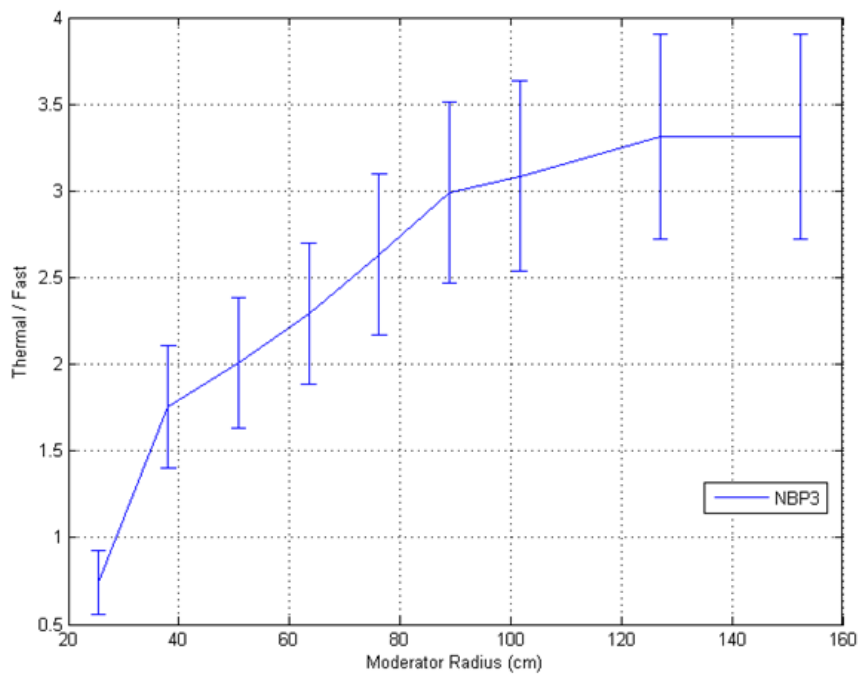


Figure 5-18: Thermal-to-fast neutron ratio as a function of moderator size in NBP3

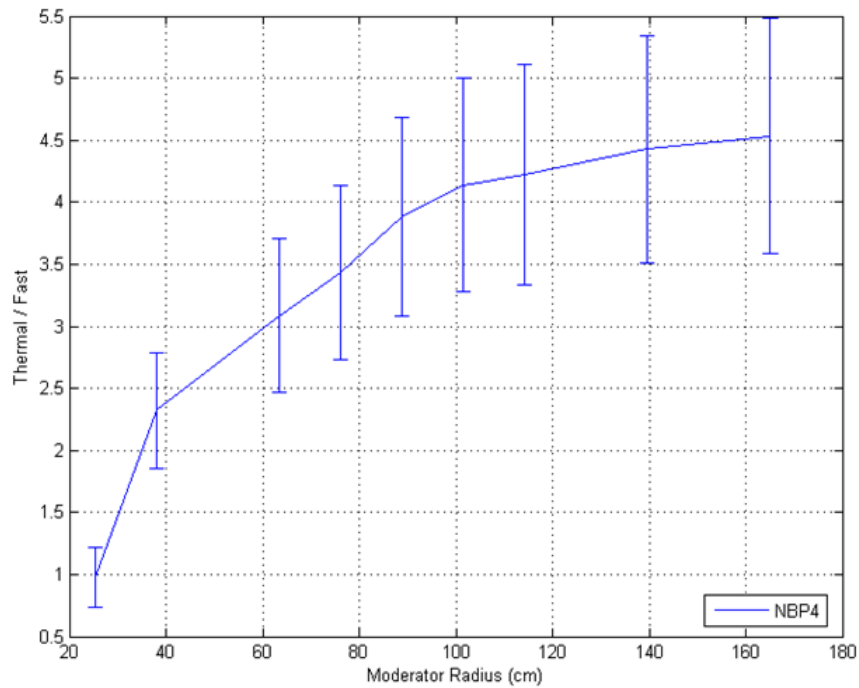


Figure 5-19: Thermal-to-fast neutron ratio as a function of moderator size in NBP4

The MCNP predicted total gamma flux at the end of NBP2, NBP3, and NBP4 are given in Table 5-6 at different moderator sizes. These results were corrected using the scaling factor given in Equation (5-11) for the gamma-rays.

Table 5-6: MCNP calculated total gamma flux on the exit surface of NBP2, NBP3 and NBP4 in the new PSBR design

Moderator Size (cm)	Total Gamma Flux ($\gamma/\text{cm}^2/\text{sec}$)					
	NBP2	Error	NBP3	Error	NBP4	Error
50.80	171.85	± 18.16	267.74	± 27.60	200.47	± 4.62
63.50	174.95	± 18.42	293.09	± 30.03	222.97	± 22.95
76.20	179.82	± 18.96	338.88	± 34.60	257.22	± 26.38
101.60	216.68	± 22.92	453.16	± 46.07	324.81	± 5.60

The MCNP simulation results showed that the thermal-to-fast neutron ratio estimated at the tally locations increases as the thermalization area enlarges around the new beam ports. However, putting more moderator material to the sides of new beam ports does not help to increase the thermal-to-fast neutron ratio. After some specific moderator radius which is specific to each new beam port, the thermal-to-fast neutron ratio stays constant. They were 76.2 cm (30 inches) for NBP2, 127 cm (50 inches) for NBP3, and 152.4 cm (60 inches) for NBP4 (See Figures 5-17, 5-18, and 5-19). The optimum radius of the moderator tank which maximizes the thermal-to-fast neutron ratio in NBP3 and NBP4 is significantly higher compared to NBP2. This was explained as follows.

NBP3 is directed to the front edge of the core and it is located between NBP2 and NBP4. On the other hand, NBP4 is almost at the tangent of the core similar to the BP4 in the existing design. It is located at the periphery of the moderator. For these beam ports, the thermalization area slowly increases as a function of moderator size. That is why a larger moderator gives better thermal-to-fast neutron ratio in these beam ports. However, the only limiting factor to install a larger moderator to the new design is the gamma component of the neutron beam. As seen from the results given in Table 5-5, the total gamma flux increases as the moderator tank enlarges. Additionally, there is a limited amount of D₂O moderator in the RSEC stock. By these competing factors, the optimum radius of the moderator tank was proposed as ~76.2 cm (30 inches). Comparison of the gamma flux spectra between the existing and new reactors with different moderator tank sizes were given In Figures 5-20, 5-21 and 5-22.

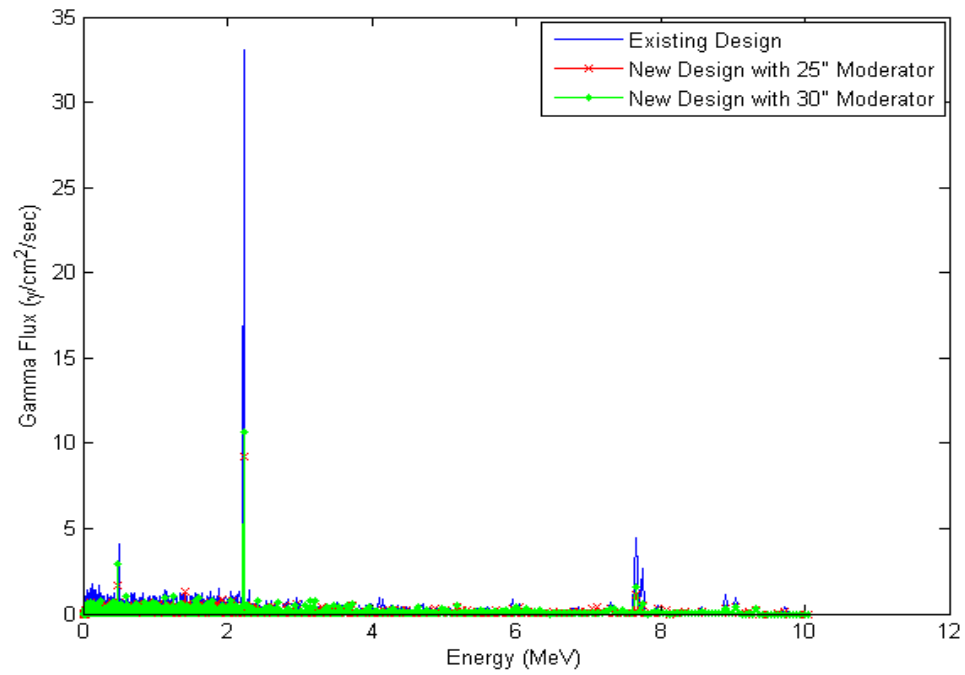


Figure 5-20: Comparison of Gamma Flux between Existing and New PSBR designs with NBP2

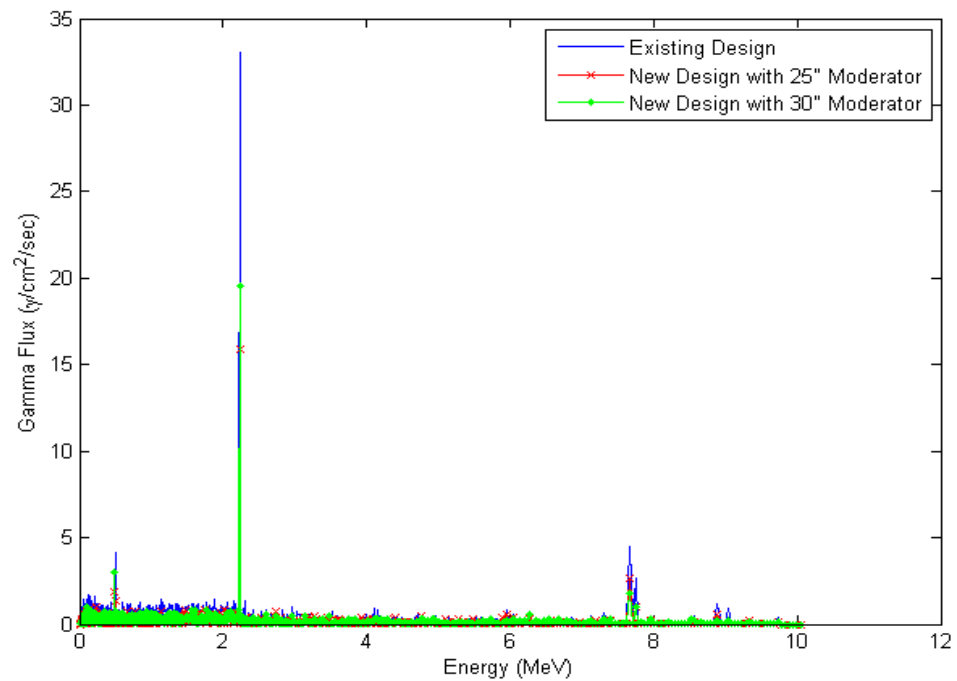


Figure 5-21: Comparison of Gamma Flux between Existing and New PSBR design with NBP3

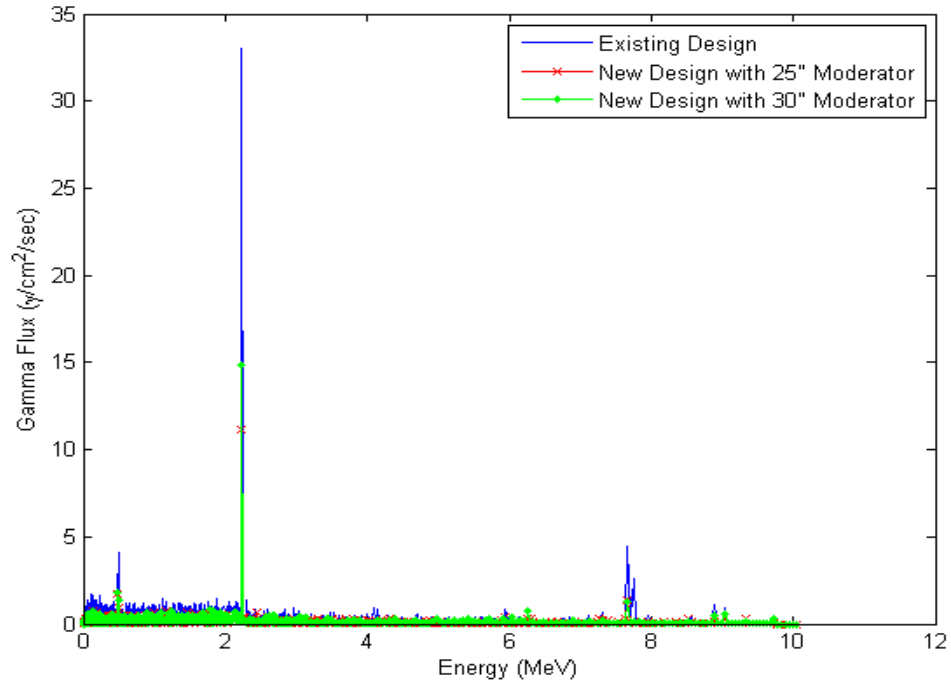


Figure 5-22: Comparison of Gamma Flux between Existing and New PSBR design with NBP4

Neutronic Performance Comparison of the New Core-Moderator Assembly with the Existing Design

The neutronic performance of the new core-moderator assembly in the optimum design was determined in this section. Beam port #4 (BP4) in the existing design was employed as reference beam port model for the new beam ports in the performance evaluation as shown in Figure 5-23. The same input parameters and assumption applied to the optimization analysis was employed in the MCNP simulation and simulation results were given only for NBP2, NBP3 and NBP4 due to two-fold symmetry in the core.

The predicted total thermal neutron flux (<1 eV) and the total fast neutron flux (>100 keV) at the end of new beam ports are given in the Table 5-7. The total gamma flux values are given in Table 5-8. The neutron and gamma flux spectra calculated at the end of new beam port are shown between Figures 5-24 and Figure 5-29.

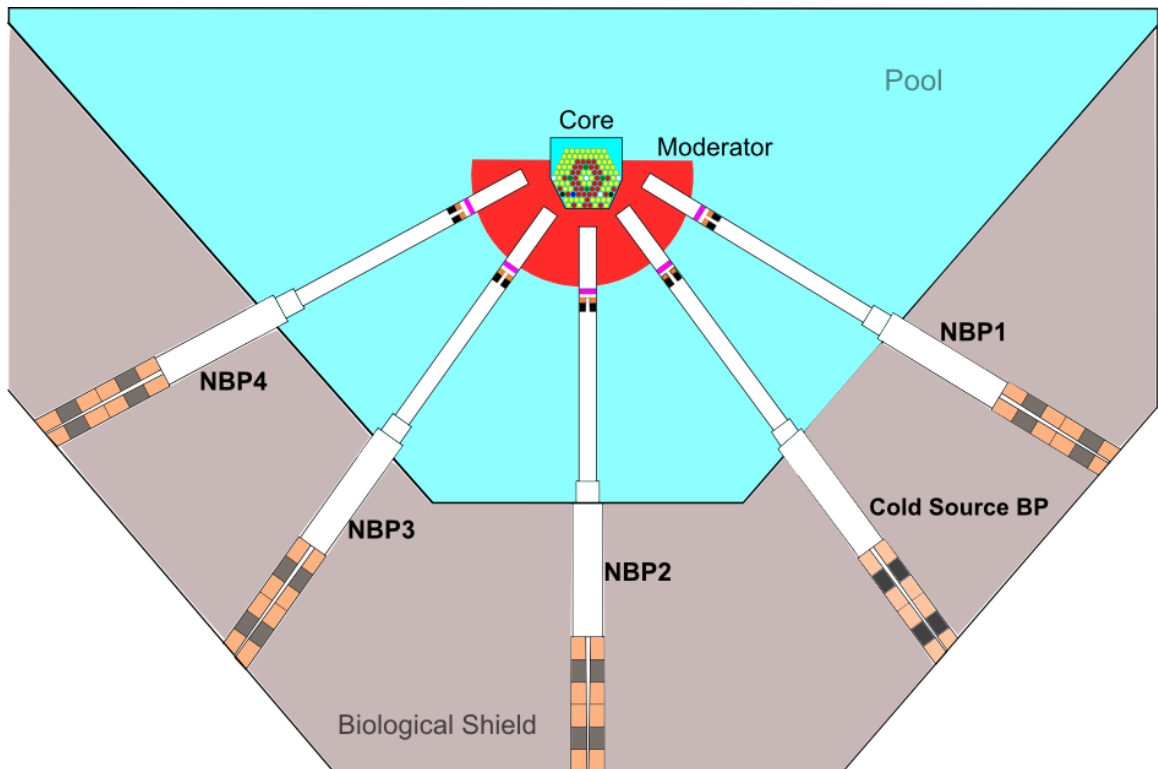


Figure 5-23: MCNP model of the new core-moderator assembly and beam ports with optimum design parameters

Table 5-7: Total thermal and total fast neutron flux calculated at the end of NBP2, NBP3 and NBP4 in the optimum PSBR design

New Beam Port	Total Thermal Flux (n/cm ² /sec)	Total Fast Flux (n/cm ² /sec)
#2	$5.29 \times 10^7 \pm 6.49 \times 10^5$	$1.70 \times 10^7 \pm 1.13 \times 10^6$
#3	$4.92 \times 10^7 \pm 9.90 \times 10^5$	$2.14 \times 10^7 \pm 4.93 \times 10^5$
#4	$4.26 \times 10^7 \pm 7.13 \times 10^5$	$7.01 \times 10^6 \pm 8.40 \times 10^5$

Table 5-8: MCNP predicted total gamma flux at the end of NBP2, NBP3 and NBP4 in the new PSBR design at 1 MW power

New Beam Port	Total Gamma Flux ($\gamma/\text{cm}^2/\text{sec}$)
#2	179.82 ± 18.96
#3	338.88 ± 34.60
#4	257.22 ± 26.38

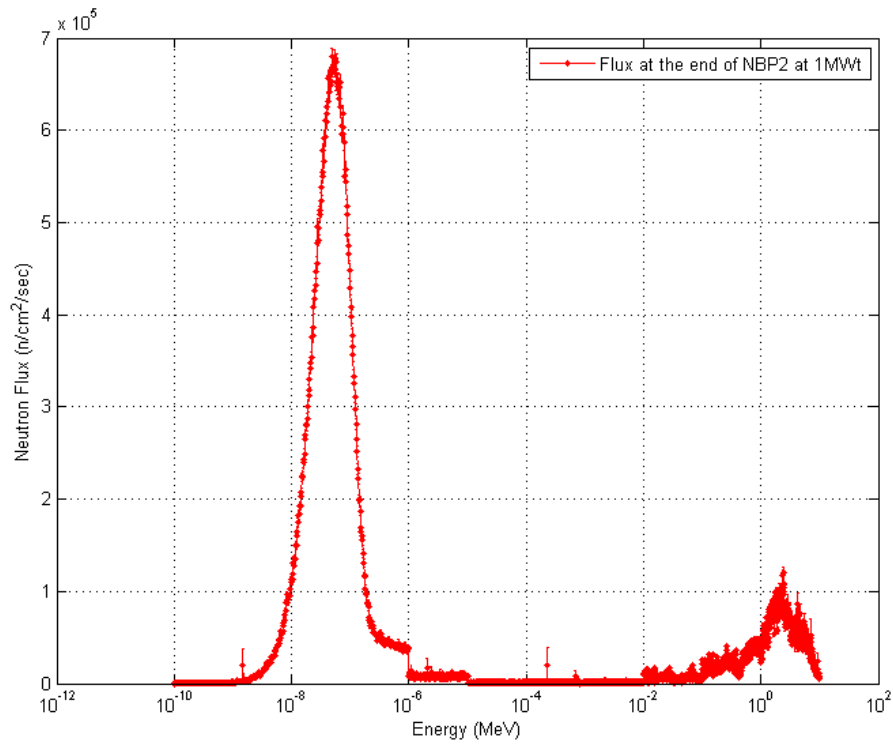


Figure 5-24: Neutron flux spectrum at the end of NBP2 in new PSBR design at 1MWt power

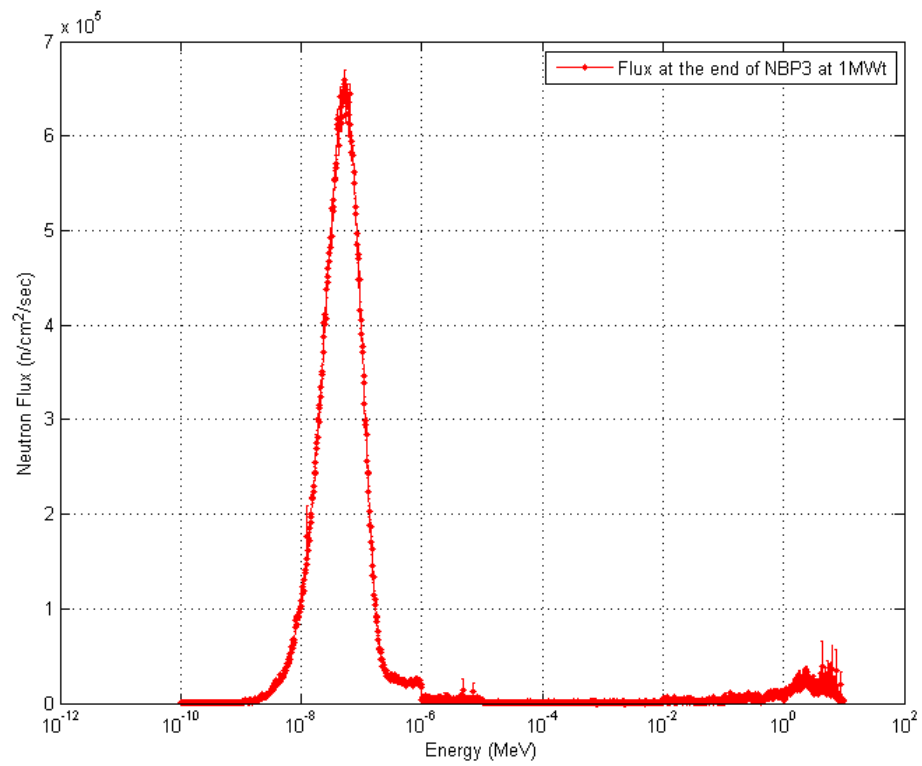


Figure 5-25: Neutron flux spectrum at the end of NBP3 in new PSBR design at 1 MW power

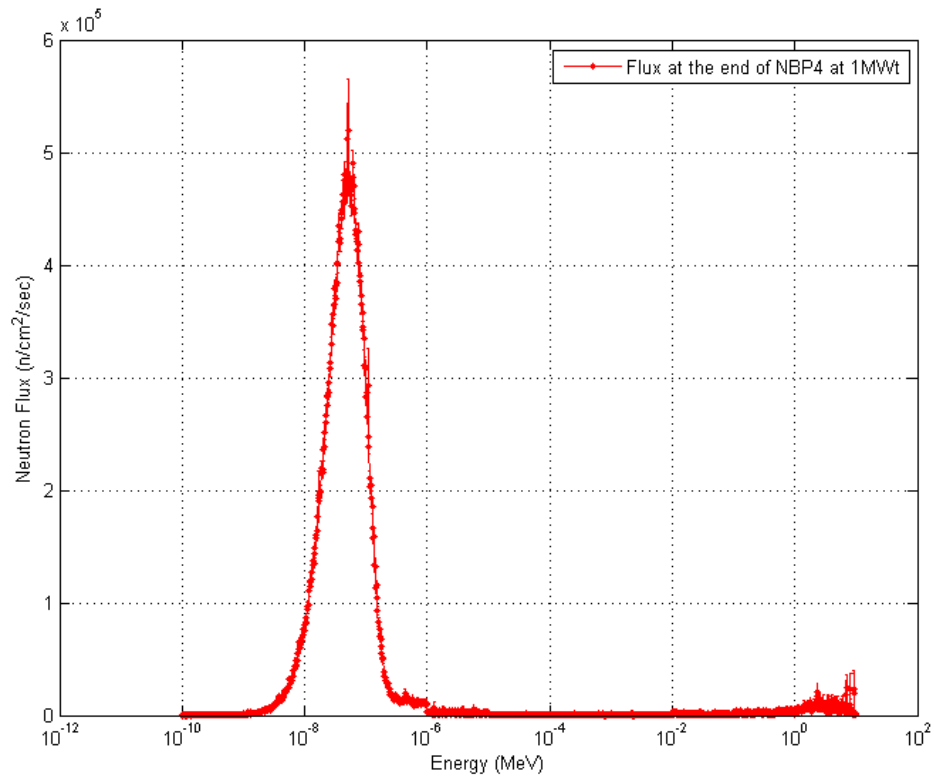


Figure 5-26: Neutron flux spectrum at the end of NBP4 in new PSBR design at 1 MWt power

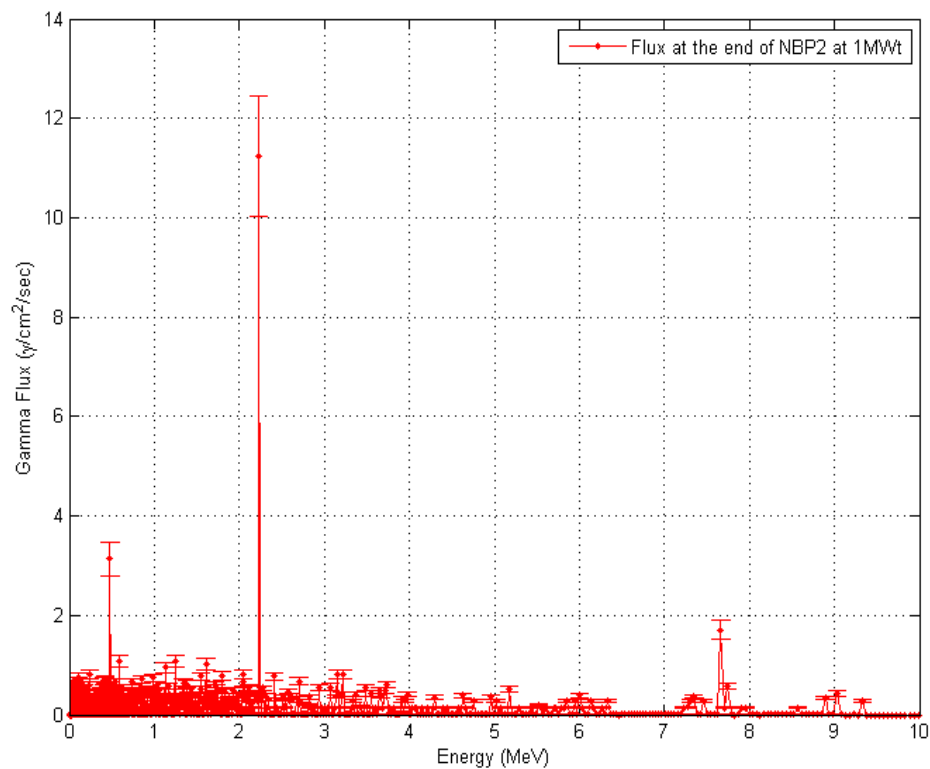


Figure 5-27: Gamma flux spectrum at the end of NBP2 in the new PSBR design with BP4 model

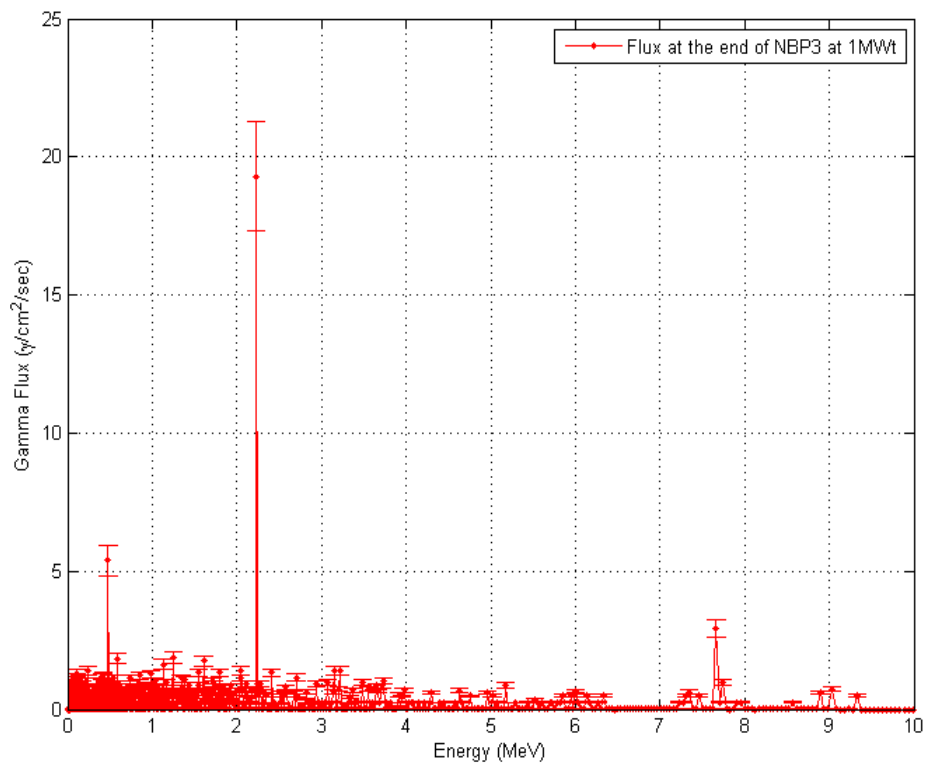


Figure 5-28: Gamma flux spectrum at the end of NBP3 in the new PSBR design with BP4 model

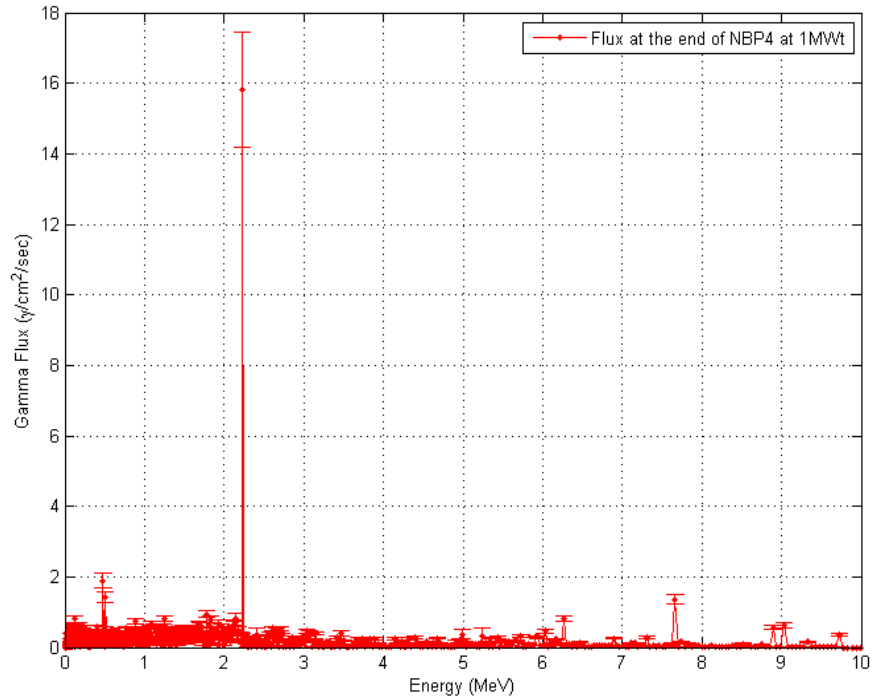


Figure 5-29: Gamma flux spectrum at the end of NBP4 in the new PSBR design with BP4 model

The estimated neutronic performance of new core-moderator assembly based on the relative increase/decrease of the total neutron and total gamma flux at the end of the new beam ports are tabulated in Tables 5-9 and 5-10. A significant performance improvement was observed based on the comparison of total thermal neutron flux between new beam ports and the existing BP4. Beside that vast majority of the gamma-rays will be eliminated in the new PSBR design as expected. However, the relative increase of the fast neutron flux in the NBP2 is significantly higher. It is very important to note here that all the new beam port models were modeled the same filter (bismuth crystal, lead and boral) and collimator designs as in the existing BP4 in the MCNP simulations. Therefore, this is a clear indication of the inefficiency of the fast neutron filtering of bismuth crystal used in existing BP4. Hence sapphire filter installation to NBP2 should be considered. This will further investigated in Chapter 7.

Table 5-9: Neutronic performance of new core-moderator assembly based on the relative increase of total thermal and total fast neutron flux in new beam ports with respect to BP4 in the existing design

New Beam Port	Total Thermal Flux Increase Rate	Error	Total Fast Flux Increase Rate	Error
#2	1.9743	±0.0460	12.6978	±2.0468
#3	1.8316	±0.0502	3.5178	±0.7312
#4	1.3507	±0.0459	1.3465	±0.1961

Table 5-10: Neutronic performance of new core-moderator assembly based on the relative decrease of total gamma flux in new beam ports with respect to BP4 in the existing design

New Beam Port	Total Gamma Flux Decrease Rate (%)	Error (%)
#2	77.03	3.36
#3	56.72	6.22
#4	67.15	4.73

Chapter 6

THERMAL-HYDRAULICS ANALYSIS of NEW CORE-MODERATOR ASSEMBLY DESIGN of the PSBR

The thermal-hydraulics performance of the existing PSBR design is based on the maximization of cross-flow among fuel rods. The cross-flow results in more coolant entering into the reactor core and improves the heat transfer behavior. However, the new design is based on optimization of moderator tank shape to improve the neutronic performance. This design change causes significant decrease in cross-flow rate along the fuel rods of the PSBR core. Thus, the thermal-hydraulics performance of the neutronically optimized core-moderator design of the PSBR needs to be analyzed to ensure the safe operation of the reactor within safety limits.

The maximum fuel temperature is the safety limitation of PSBR core both in normal operation and Loss of Coolant Accident (LOCA) conditions. The fuel temperature should not exceed 1150 °C in steady state and 950 °C in LOCA. In both conditions, cladding outer temperature is expected to be 500 °C. The safety limit aims to prevent hydrogen release from the TRIGA fuel which causes stress on the cladding and potential rupture in case the ultimate strength of the cladding is exceeded. The reactor core is cooled with natural convection under steady state conditions. Figure 6-1 shows the hypothetical flow field with axial and cross-flow paths in and around the existing PSBR core and the expected flow paths in the new design is given in Figures 6-2 and 6-3, respectively. The flow is driven by density and hydrostatic pressure difference along fuel rods in natural convection. The subcooled nucleate boiling near cladding surface is also expected when the water exceed the saturation temperature at the outer surface of cladding [24]. However, the condition of bubble nucleation depends on the wall superheat and the roughness of the claddings' outer surface. The expected PSBR flow condition consists of both

natural convection driven flow and subcooled boiling. Under these conditions, the measured fuel temperature in instrument rod-16 (I-16) is 540 °C which is below the safety limit.

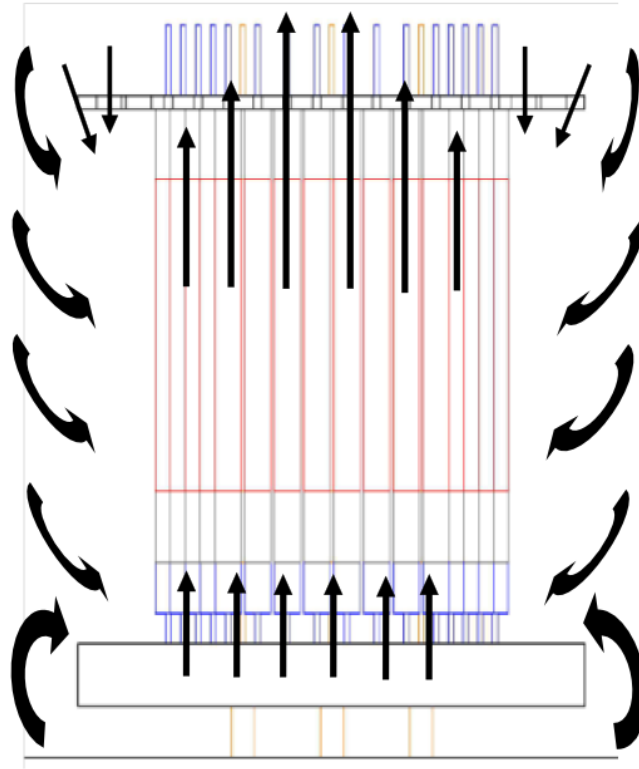


Figure 6-1: The hypothetical flow field in and around the existing PSBR core

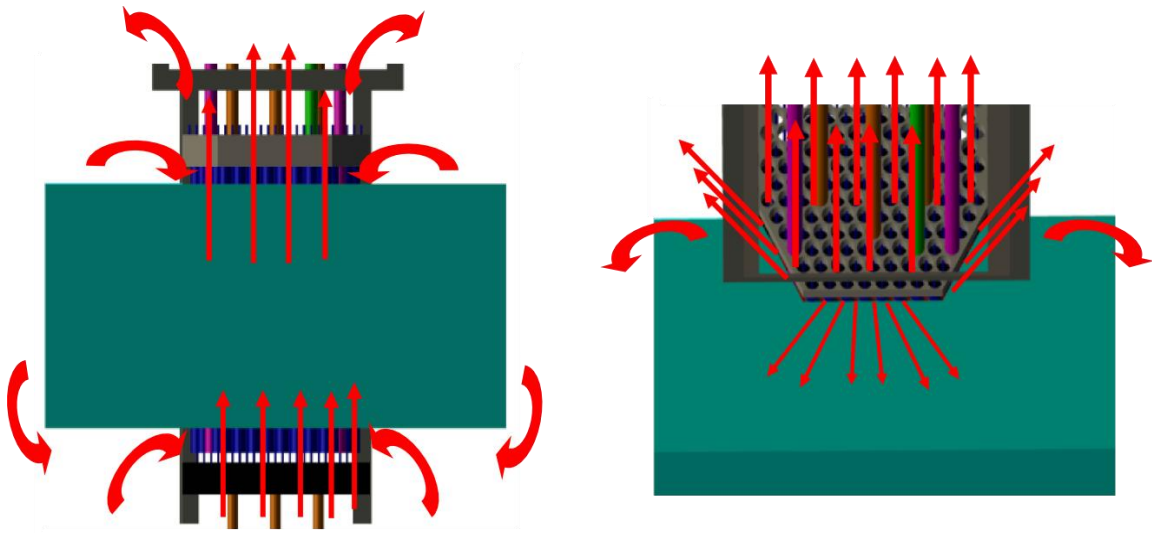


Figure 6-2: The flow path in and around the new core-moderator assembly design

The moderator tank in the new design decreases the cross-flow entering into the core and alters the flow geometry of fuel rods on the core periphery. These changes may adversely affect the heat transfer from cladding to the bulk fluid by increasing the bubble nucleation rate. As a result, subcooled nucleate boiling regime may be shifted to nucleate boiling regime in which the bulk coolant temperature is at saturation. Then, the fuel temperature may rise due to the increment on the cladding outer surface temperature. Hence, thermal-hydraulics performance under steady state conditions for the PSBR core is modeled by computational fluid dynamics (CFD) approach. There are three approaches that can be applied to this problem; utilization of sub-channel analysis codes like COBRA-TF [28], modeling with porous media approach [24], and using conventional CFD by modeling all flow details in the system. Subchannel analysis codes are very powerful for conventional nuclear power plants and it is possible to modify this approach for research reactors if flow is not perturbed by complex moderator tank design. Porous media approach is also possible to be utilized but the main drawback is the accurate definition of momentum and heat transfer sources by considering axial flow change from laminar to turbulent and the boiling. In this study, a conventional CFD method is employed for thermal-hydraulics performance evaluation of the new PSBR core-moderator assembly design. Although this technique is computationally expensive, it is possible to model complex model geometries of the top and bottom grid plates. Also, if a fine computational mesh exists similar to the used model in this study, it is easy to include the effect of thermal and velocity boundary layers to the thermal-hydraulics calculation.

CFD Modeling of the New PSBR Design

CFD modeling of the new PSBR design was performed in the ANSYS Fluent v14.0 environment which provides simultaneous computation of both density driven flow and the two-phase momentum and heat transfer equations. The ANSYS Fluent code utilizes Finite Volume Method (FVM) to solve momentum, energy, and associated turbulence equations for both liquid and vapor phases. The FVM is based on the conservation of mass, momentum, and energy in a given computational cell. Although core geometry has a two-fold symmetry, the burnup of the fuel rods are unfortunately not symmetric. Hence, each fuel rod has a different heat generation rate determined by the burnup. The heat generation rates of fuel rods are estimated by TRIGSIMS burnup code [17] which periodically calculates the depleted fuel compositions at five axial sections of the fuel rods since 1965.

There are two possible CFD modeling approaches in fuel rod-bundle systems: (1) the conjugate heat transfer approach which solves the fluid and solid equations simultaneously by coupling the heat transfer rate from fuel region to coolant. In this approach, each individual fuel rod including clad, gap, and fuel is modeled. The major disadvantage of this method for our case study is the excessive number of fuel rods, which increases the number of computational cells. Furthermore, the heat transfer coefficient of the gap as a function of burnup is also not available even though the thermal properties can be assumed constant, not a function of burnup. Since the PSBR is in steady state conditions, the second approach (2) is performed on CFD modeling as follows. The amount of heat generated by fuel is assumed to be transferred into the coolant. Therefore, surface heat flux is applied to cladding outer surface by considering the burnup level of the individual fuel rod in reference core loading 53H at 1MW full power. Then, conservation equations are solved for only coolant (water and vapor). This approach is reasonable since the safety limit of the PSBR is defined under steady state conditions.

The burnup dependent heat flux values at the surface of the fuel rod claddings (q'') are calculated by using the burnup dependent local power with the following relation:

$$q'' \left[\frac{W}{m^2} \right] = \frac{\dot{Q} [W]}{2\pi r_c H}, \quad (6-1)$$

where r_c is the outer surface radius of clad (1.83 cm) and H is the height of each fuel rod section in the active fuel length (7.62 cm).

Geometry and Mesh Structures generated in the CFD Model of the New PSBR Design

Initially, the core-moderator assembly was drafted in AutoCAD. Then, the computational mesh was generated using the ANSYS Gambit mesh generator. Two different meshes were utilized in the model. Namely structured rectangular cells were used in the core region and unstructured tetragonal cells in the pool, shown in Figures 6-3, 6-4, 6-5, and 6-6. The number of mesh is 22 million for core and pool regions. This mesh structure provided the accuracy and convergence in the simulation results. The meshing and CFD simulation were performed on the Lion-lsp cluster, a HPC cluster of Pennsylvania State University. To examine the mesh dependence of the solution, eight subchannels with 10 fuel rods around the centre thimble were analyzed first. The mesh quality was changed until all the solution parameters become independent from the mesh quality. Then this mesh was applied to the CFD model of the new PSBR design. Using this methodology, a mesh independent solution was achieved.

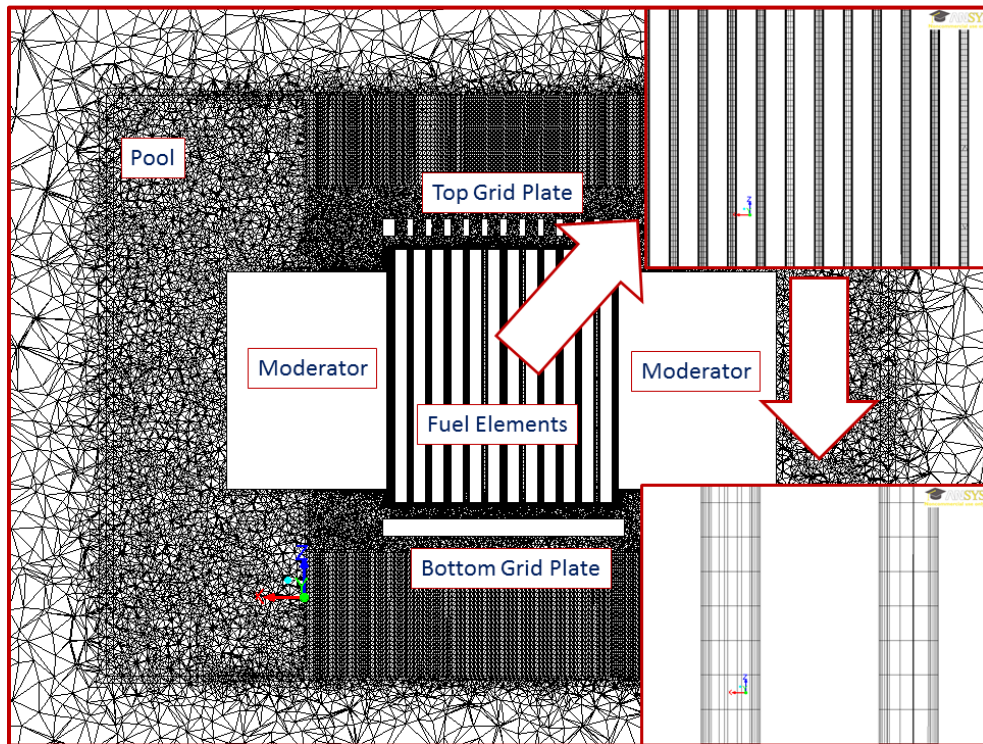


Figure 6-3: Computational mesh (side view) in the core-moderator assembly and the pool

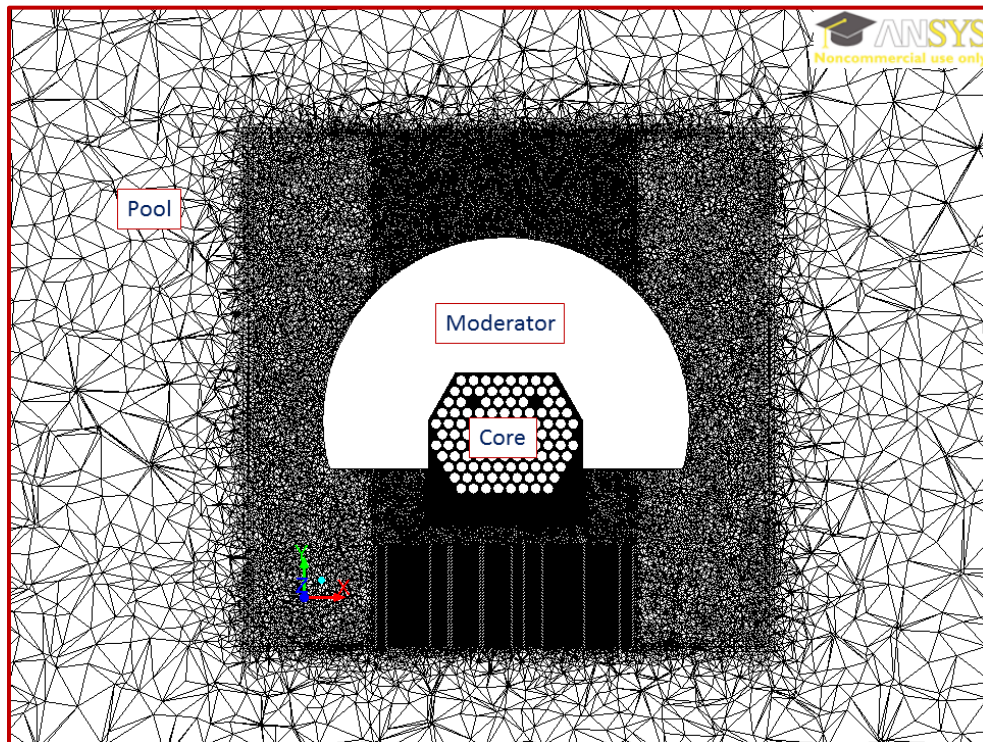


Figure 6-4: Computational mesh (top view) in the core-moderator assembly and the pool

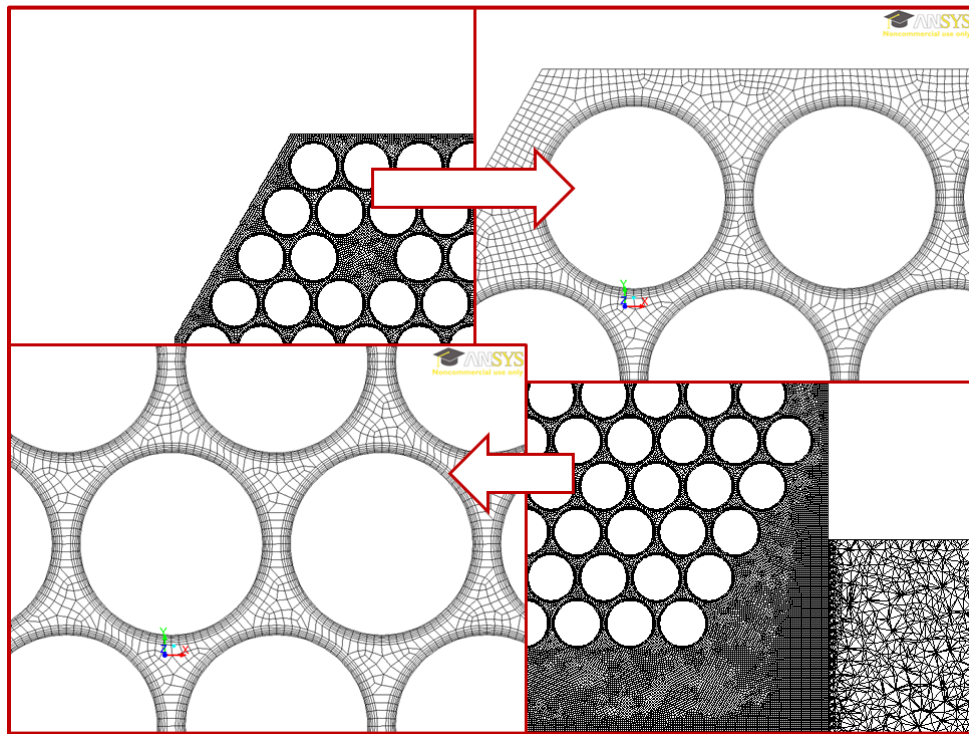


Figure 6-5: Computational mesh in the core (top view)

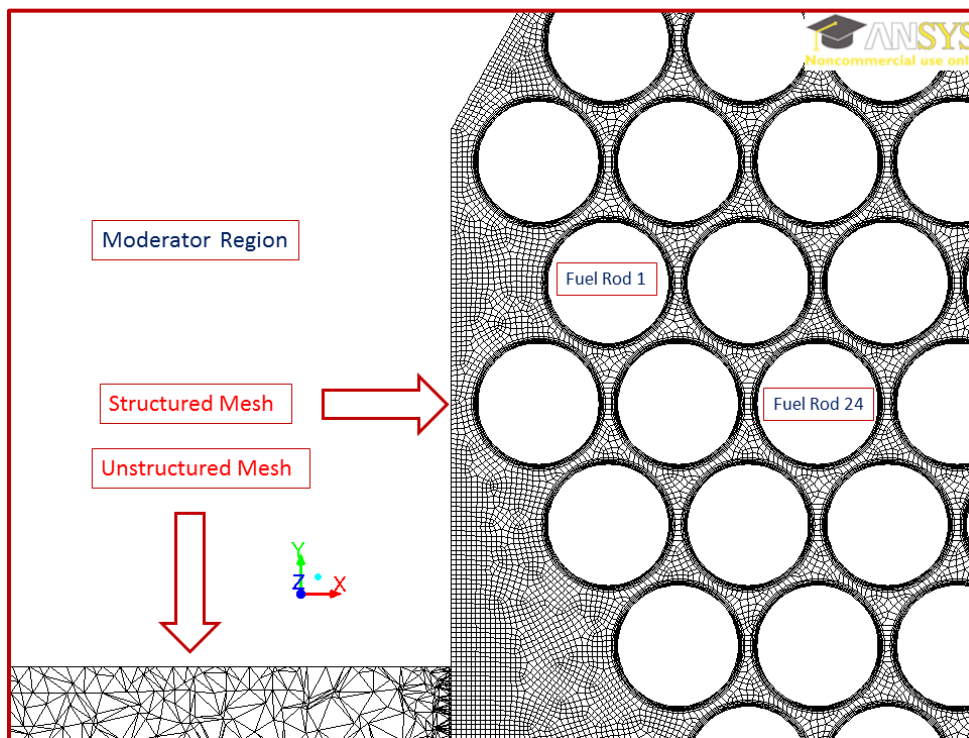


Figure 6-6: Computational mesh at the periphery of the core (top view)

Physical Models used in the CFD Modeling of the New PSBR Design

Heat Transfer Mechanisms from Clad Outer Surface to Coolant in the PSBR

The heat is transferred from the fuel rod to coolant via thermal conduction in the boundary layer of the coolant, the natural convection of bulk coolant and the subcooled nucleate boiling of the coolant at the clad surface. The thermal conduction occurs in the thermal boundary layer as

$$q'' = -k_{coolant} \frac{\partial T}{\partial \vec{r}} \quad (6-2)$$

The natural convection is driven by the local coolants' density difference which is a function of temperature. The density difference due to temperature gradient is determined by the thermal expansion coefficient β and it can be written as

$$\beta = -\frac{1}{\rho} \left(\frac{\partial \rho(T)}{\partial T} \right)_p \quad (6-3)$$

The difference in density as a result of temperature gradient along the fuel channel causes the buoyancy force acting on the coolant. This buoyancy force creates an additional component in the momentum and the energy equations, which promotes convection via local density change. The last heat transfer mechanism is the phase transition *aka* vaporization. When the saturation temperature is reached at the cladding outer surface, bubbles are likely to nucleate on the cladding surface. The bubble creation near the wall requires sufficient surface tension forces to sustain. Therefore, the bubble nucleation is a strong function of the claddings surface roughness.

Flow Conditions in the Coolant Channels of the PSBR

The momentum, in general, is transferred due to diffusion and convection. When the momentum due to the diffusion is higher than the momentum of the convection, the flow is dubbed as laminar. The opposite case is called turbulent. In forced convection, the Reynolds (Re) number determines whether flow is laminar or turbulent. On the other hand, the Rayleigh (Ra) number determines the flow condition in natural convection. Rayleigh number is expressed as

$$Ra = \frac{\vec{g}\beta\Delta TL^3\rho}{\mu\alpha} \quad (6-4)$$

The critical Rayleigh number from laminar to turbulent for heated vertical plate is 10^9 . In our fuel rod bundle configuration, the estimated Rayleigh number is 7×10^9 by the CFD simulation of the new PSBR design. Therefore, buoyancy driven flow in coolant channels can be assumed as turbulent. This assumption is not valid at the lower parts of fuel rods where there is no heat generation. However, the turbulent model, $\kappa - \omega$ model with SST formulation [59], that is utilized in the calculation has good accuracy in laminar to turbulence transition.

Multiphase Modeling of the PSBR in ANSYS Fluent

Under circumstances where the saturation temperature of coolant is exceeded at the claddings outer surface, bubble nucleation will initiate and this will provide sufficient wall superheat for the following subcooled nucleate boiling. Therefore, the CFD calculation is bound to have multiphase flow modeling. There are three basic multiphase flow models utilized in the nuclear industry. First one is the homogeneous equilibrium model (HEM) in which phase velocities and temperatures are assumed to be same [60]. This approach has a good correlation at high mass fluxes and high operating pressures. Another approach, Drift Flux Model (DFM) is a modification to HEM flow by including the drift between phases [60]. ANSYS Fluent provides a

mixture formulation that can be used to mimic both methods. The third one is the two-fluid model which is based on Eulerian-Eulerian formulation of phases. In this method, conservation of momentum and energy equations are solved separately for liquid and vapor phases. Both liquid and vapor phases are treated as incompressible and Newtonian fluids. Single phase and Eulerian-Eulerian multiphase formulations are given in Appendix A and Appendix B, respectively. All methods assume that the second phase has already existed in the flow. However, bubbles as second phase are created in the PSBR operating conditions. Therefore, existing approaches are not valid to model without bubble nucleation models. ANSYS Fluent also provides an additional option to simulate boiling in Eulerian-Eulerian method. Thus, it is possible to simulate thermal-hydraulics performance of the new PSBR design by using Eulerian-Eulerian multiphase approach with Rensselaer Polytechnic Institute (RPI) wall boiling model [61].

The heat flux, q'' from wall to two phase fluid in RPI model is partitioned as

$$q'' = q''_{1\phi} + q''_q + q''_e. \quad (6-5)$$

In Equation (6-5), $q''_{1\phi}$ is the single-phase convection heat flux, q''_q is denoted as quenching heat flux from the wall to bulk fluid that fills the volume vacated by the bubbles, and the q''_e stands for the wall evaporation heat flux that is used to generate vapor bubbles. Figure 6-7 highlights the subcooled flow regime which is observed in the PSBR nominal operating conditions. The PSBR core is operating with isolated bubbles at low pressure, which is ~ 1.5 atm. When the wall temperature sufficiently rises above the local saturation temperature, pre-existing vapor on the wall can nucleate and grow. This temperature, T_{ONB} indicates the onset of subcooled boiling (ONB). The thickness of the bubble layer starts increasing at the onset of significant boiling (OSB). Subcooled boiling region at low pressure can be divided into two major regions, “highly subcooled” and “low subcooled” regions [62]. At the low subcooled region, a thin, long and almost flat film layer with 5 to 10% void fraction forms at low pressures [63]. At highly

subcooled boiling region, the void fraction and thickness of the film layer significantly increase. As more energy is transferred into the liquid, these bubbles can grow and detach from the surface. However, the majority of the detached bubbles condense in the subcooled liquid. The temperature difference between the vapor and subcooled liquid causes relative drift between the phases.

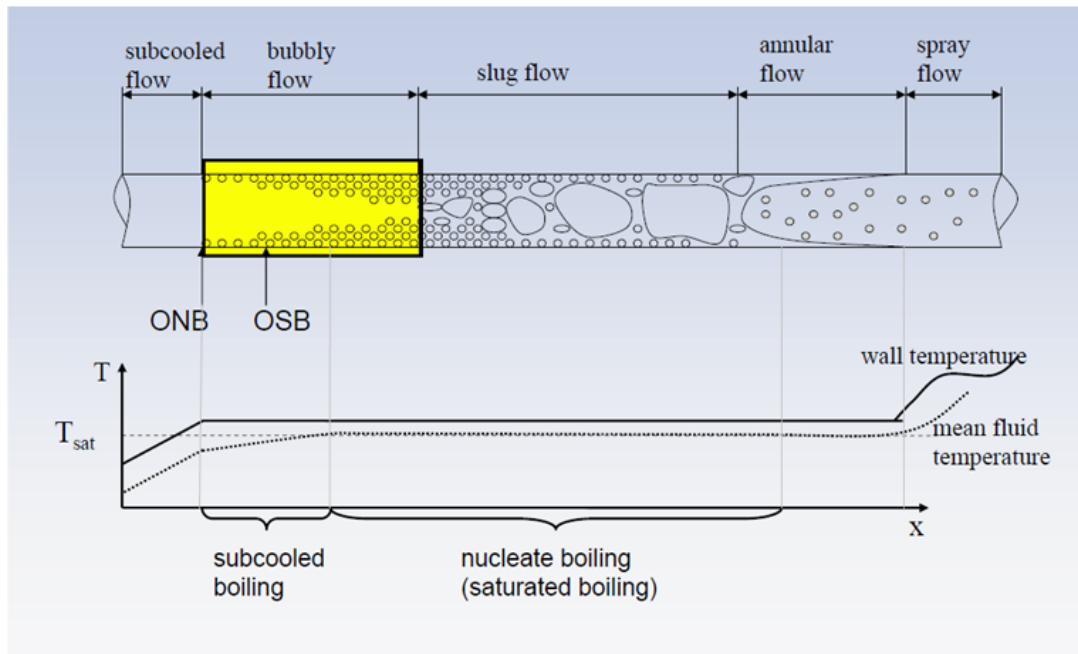


Figure 6-7: Multiphase flow regimes in boiling water flow. Pre-existing bubbles on the wall sites nucleate and grow at Onset of Nucleate Boiling (ONB). The thickness of the bubble layer increases starting from the Onset of Significant Boiling (OSB) [64]

RPI Wall Boiling Model

Rensselaer Polytechnic Institute (RPI) wall boiling model was developed by Kurul and Podowski [61]. As given before in Equation (6-5), the wall heat flux from wall to subcooled flow is partitioned into three components: single-phase convective heat flux, the quenching heat flux, and the evaporative heat flux. In the RPI model, the quenching heat flux is modeled as

$$q''_Q = \frac{2k_l}{\sqrt{\pi\lambda_l T}} (T_w - T_l), \quad (6-6)$$

where k_l is the conductivity of the liquid phase, T is the periodic time, λ_l is the diffusivity, T_w is the wall temperature, and T_l is the liquid temperature. The diffusivity in this equation can be defined as

$$\lambda_l = \frac{k_l}{\rho_l C_{pl}}. \quad (6-7)$$

The evaporative heat flux is modeled in the RPI model with the following relation.

$$q''_e = V_d N_W \rho_v h_{fv} f, \quad (6-8)$$

where V_d is the volume of the bubbles at the departure diameter (D_W), N_W is the nucleate site density, ρ_v is the vapor density, h_{fv} is the latent heat of evaporation, and f is the bubble departure frequency.

The single-phase convective heat flux is modeled as

$$q''_{1\phi} = h_c (T_w - T_l) (1 - A_b), \quad (6-9)$$

In this equation, A_b stands for the bubble influence area which defines the area covered by the bubbles on the heated wall. In the RPI model, the bubble influence area is calculated as

$$A_b = K \frac{N_W \pi D_W^2}{4}, \quad (6-10)$$

where K is an empirical constant which varies 1.8 and 5. In this study, an empirical constant based on Del Valle and Kennings study [65] was employed for K .

Bubble departure frequency used in the evaporative heat flux is calculated using the following relation [66].

$$f = \frac{1}{T} = \sqrt{\frac{4g(\rho_l - \rho_v)}{3\rho_l D_W}}. \quad (6-11)$$

In Equation (6-10), bubble departure diameter D_W (mm) and nucleation site density N_W are based on the empirical correlations [67][68] given as

$$D_W = \min\left(0.0014, 0.0006e^{\frac{-\Delta T_W}{45}}\right), \quad (6-12)$$

and

$$N_W^* = f(\rho^*)r_c^{*4.4}. \quad (6-13)$$

In Equation (6-13),

$$N_W^* = C^n (T_W - T_{sat})^n D_W^2, \quad (6-14)$$

$$r_c^* = \frac{4\sigma T_{sat}}{\rho_v h_{fv} \Delta T_w D_w}, \quad (6-15)$$

$$f(\rho^*) = 2.157 \times 10^{-7} \left[\frac{\rho_l - \rho_v}{\rho_v} \right]^{-3.2} \left\{ 1 + 0.0049 \left[\frac{\rho_l - \rho_v}{\rho_v} \right] \right\}^{4.13}. \quad (6-16)$$

The complete subcooled boiling method highly depends on accurate modeling of two phase interfacial mass, momentum, and heat transfer terms which are given in Appendix B for two fluid method. The accurate representation can be given by modeling the interfacial area concentration. The void fraction and the interfacial area represent the first order geometrical parameters. Thus, the shape of the nucleated bubble is a strong function of the interfacial area.

The Modeling of Interfacial Area Concentration in the ANSYS Fluent

Interfacial area concentration defines the area between the phases per unit mixture volume. The accuracy of predicting the momentum, heat and mass transfer in the interface depends on the correct formulation of interface area between the phases.

The transport equation for the interfacial area concentration can be written as

$$\frac{\partial(\rho_g \chi_p)}{\partial t} + \nabla \cdot (\rho_g \bar{u}_g \chi_p) = \frac{1}{3} \frac{D\rho_g}{Dt} \chi_p + \frac{2}{3} \frac{\dot{m}_g}{a_g} \chi_p + \rho_g (S_{RC} + S_{WE} + S_{TI}), \quad (6-17)$$

where χ_p is the interfacial area concentration, $\dot{m}_g$ is the mass transfer rate into gas phase per unit mixture volume, S_{RC} is the coalescence sink term, S_{WE} is the wake entrainment term, S_{TI} is the

breakage source term, and a_g is the gas volume fraction. There are three sets of models, Hibiki-Ishii model [69], Ishii-Kim model [70] and Yao-Morel model [71], available in Fluent code for the source and sink terms.

In the CFD analysis, coalescence and breakage kernels were defined by Ishii-Kim model, nucleation rate was defined with Yao-Morel model (critical Weber number = 6), Wu-Ishii-Kim model was employed for dissipation rate, and the diameter of the bubbles were defined with Sauter-mean diameter [72]. Detailed information about these models are given in reference [73].

Results and Discussion

In this section, the CFD simulation results were compared with the experimental measurements and Cobra-TF (CTF) simulation results in the existing PSBR design [28]. The thermal-hydraulics performance of the new PSBR design was estimated by two phase formulation and the bulk fluid temperature measured at the center of hot channel (Channel-R) in the core loading of 53H. The measurements were performed using thermocouples inserted into a long aluminum tube with 0.6-cm outer diameter. The location of the thermocouple in the existing PSBR core is the Channel-R as shown in Figure 6-8. In Channel-R, the bulk fluid temperature was measured in two axial locations: (1) center of the core where the heat flux from the fuel rod is maximum and (2) 12.7 cm above the core center. The simulation results were compared to the measurements at these locations in Channel-R as well as to Cobra-TF (CTF) simulation results in the existing PSBR design. Figure 6-9 shows the local power distribution in the PSBR core for 1 MW power operation of the reactor.

The radial flow paths between fuel rods at the center of the core are shown in Figure 6-10. The new moderator geometry results in secondary flow entering due to decrease in the cross-flow area. Notably, the radial flow improves the local heat transfer conditions by providing

radial-mixing in the core. Figure 6-11 and 6-12 show the overall axial flow redistribution along the coolant channels where reasonable flow symmetry has been observed. Focusing on the bottom grid plate, orifice effect is seen in Figure 6-13. The main observation according to flow paths, the amount of fluid which is fed to the channels is a function of rod location. Another observation is the flow development along the channels. Due to the different local heat transfer conditions which are function of heating on the fuel rods, flow develops from laminar velocity profile to turbulent as seen in Figure 6-14.

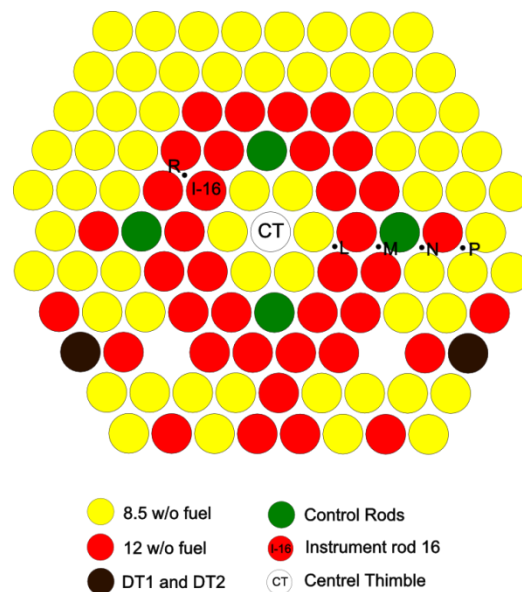


Figure 6-8: Locations of flow channels where axial coolant temperature profiles are compared between measurement and simulations in new PSBR core

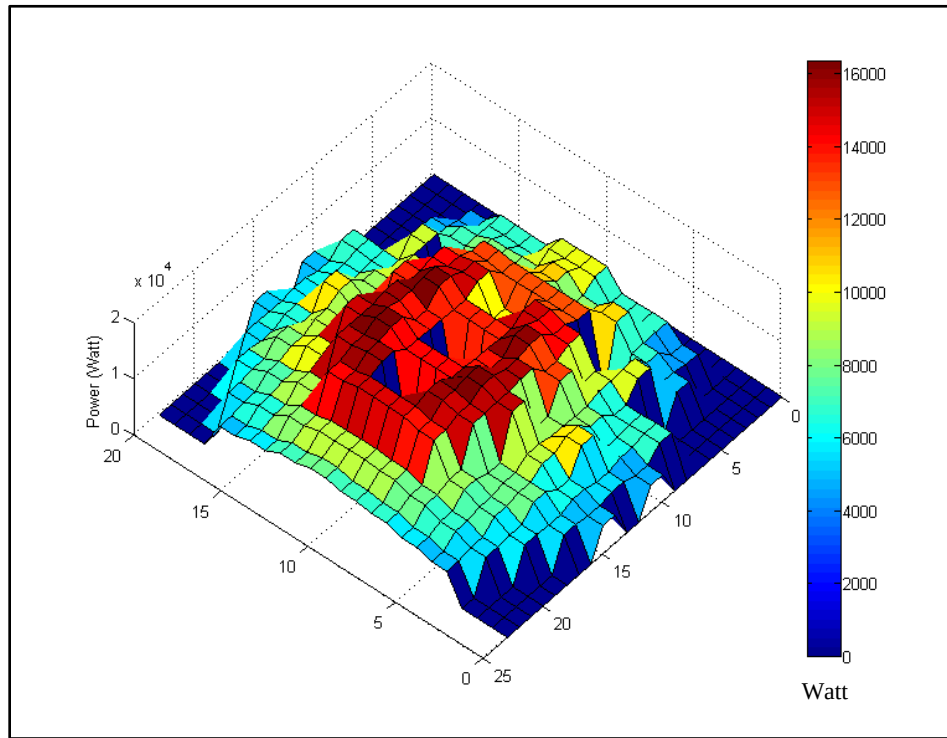


Figure 6-9: The local power distribution in the PSBR core for 1 MW power operation of the reactor

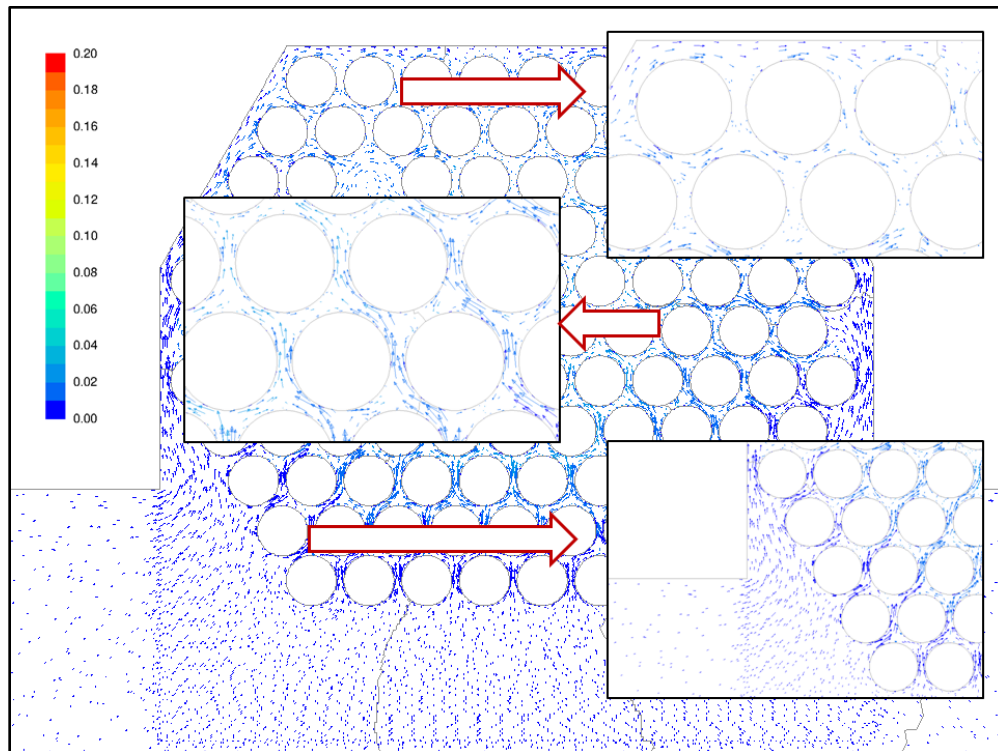


Figure 6-10: Radial flow paths colored by velocity magnitude (m/s) at the center of the core

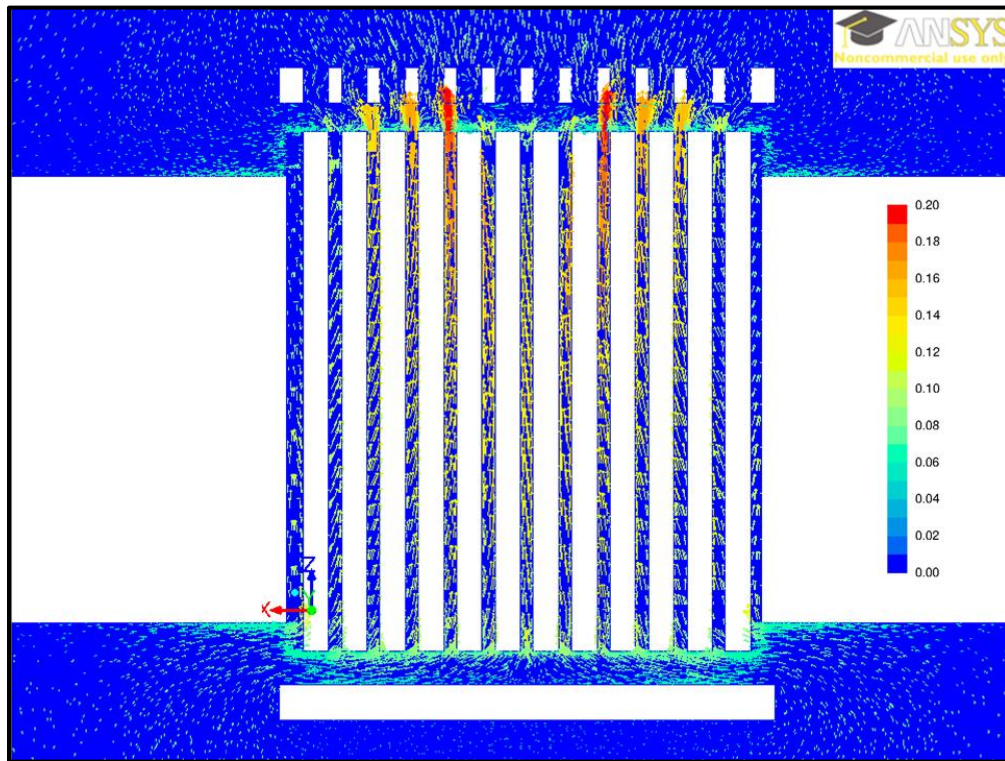


Figure 6-11: Axial flow paths colored by velocity magnitude (m/s) along the fuel rods in the core

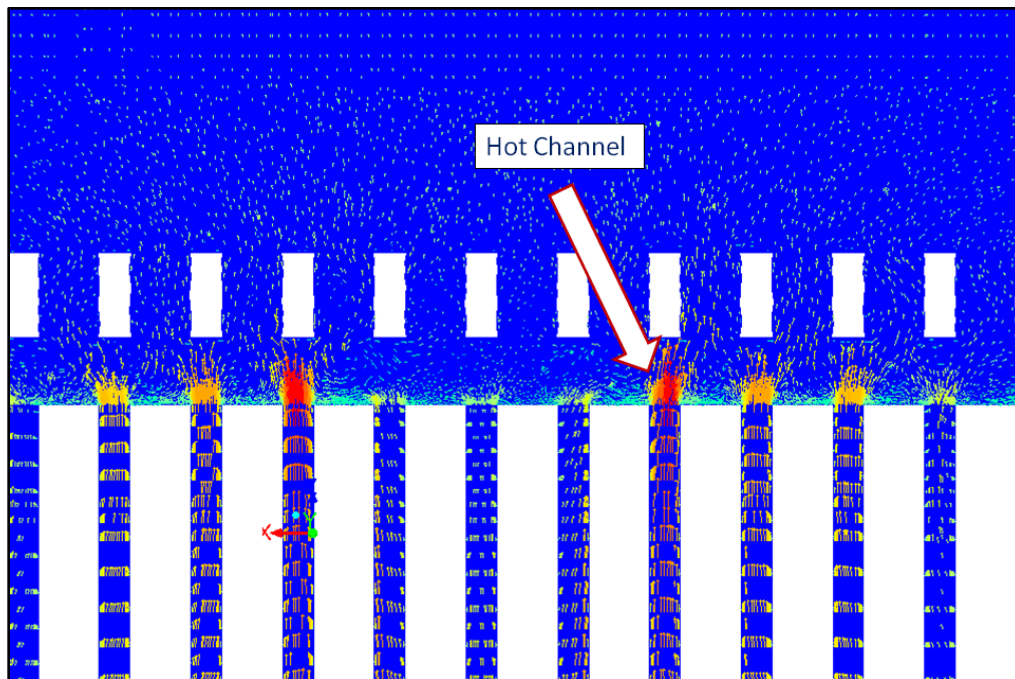


Figure 6-12: Flow paths colored by velocity magnitude (m/s) in the region between the fuel rods and the top grid plate

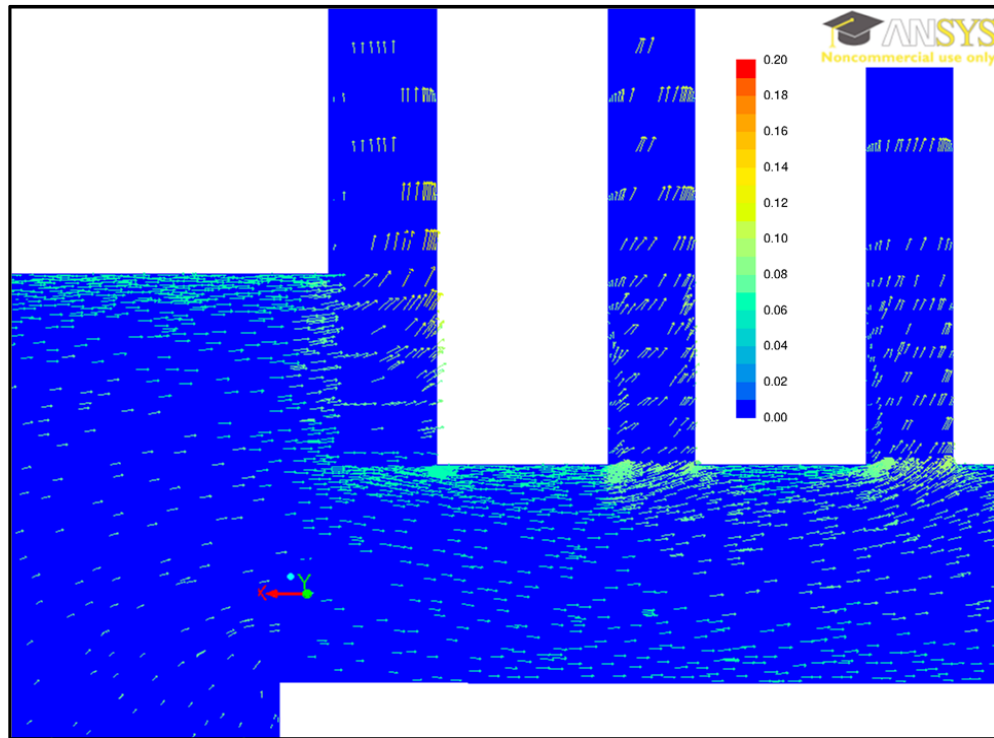


Figure 6-13: Flow paths colored by velocity magnitude (m/s) in the region between the fuel rods and the bottom grid plate

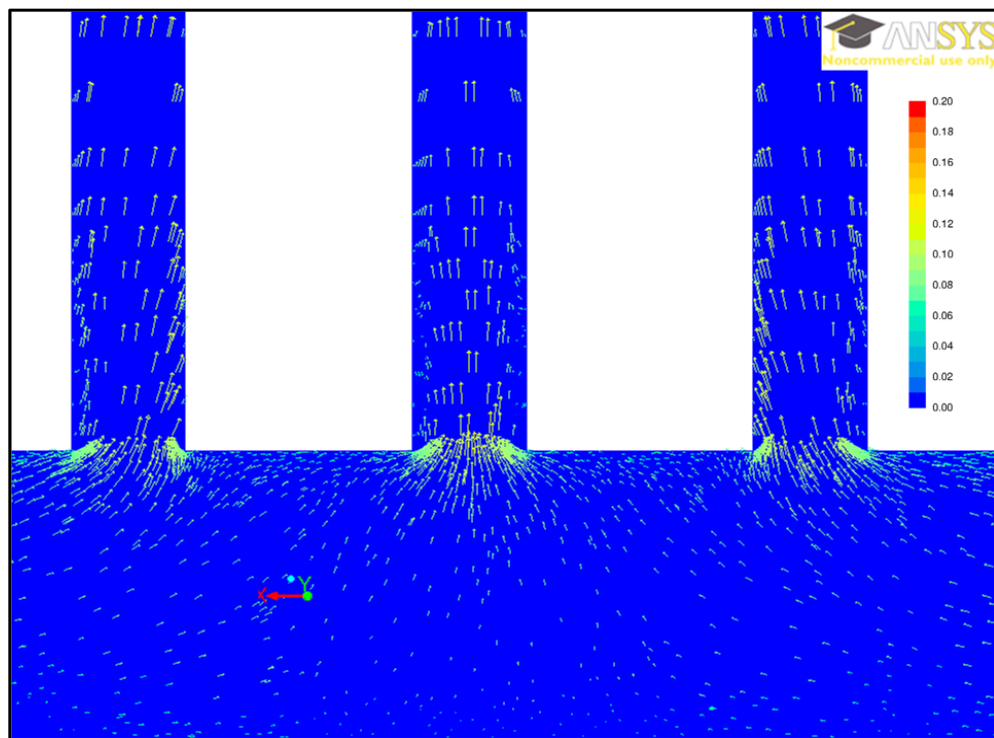


Figure 6-14: Flow paths colored by velocity magnitude (m/s) at the entrance of the flow channels

The cross-flow profile between the fuel rods at the center of the core is shown in Figure 6-15. The cross-flow is entering into the core from the pool side and is symmetrically distributed between the fuel rods and along the core-moderator interface. Figure 6-16 shows the cross sectional view of bulk fluid velocity distribution (as contours) in the estimated hot channel and at the center of the core. The calculated velocity varies between 0.12 and 0.14 m/s. There are two spots where the maximum velocity has been reached (see Figure 6-12). At these locations the bulk temperature increases up to 88 °C which is the maximum temperature in the new PSBR core. Besides, both Figures 6-17 and 6-18 show that the velocity and thermal boundaries are reasonably well defined. Contours of void fraction on the heated fuel rods in the hot channel are given in Figure 6-18. This figure shows the bubbles forming on the fuel rods. Since no bubble was observed in the bulk fluid, bubbles are condensing in the thermal boundary layer before reaching to bulk fluid which is subcooled. Therefore, bubbles do not change the bulk fluid properties.

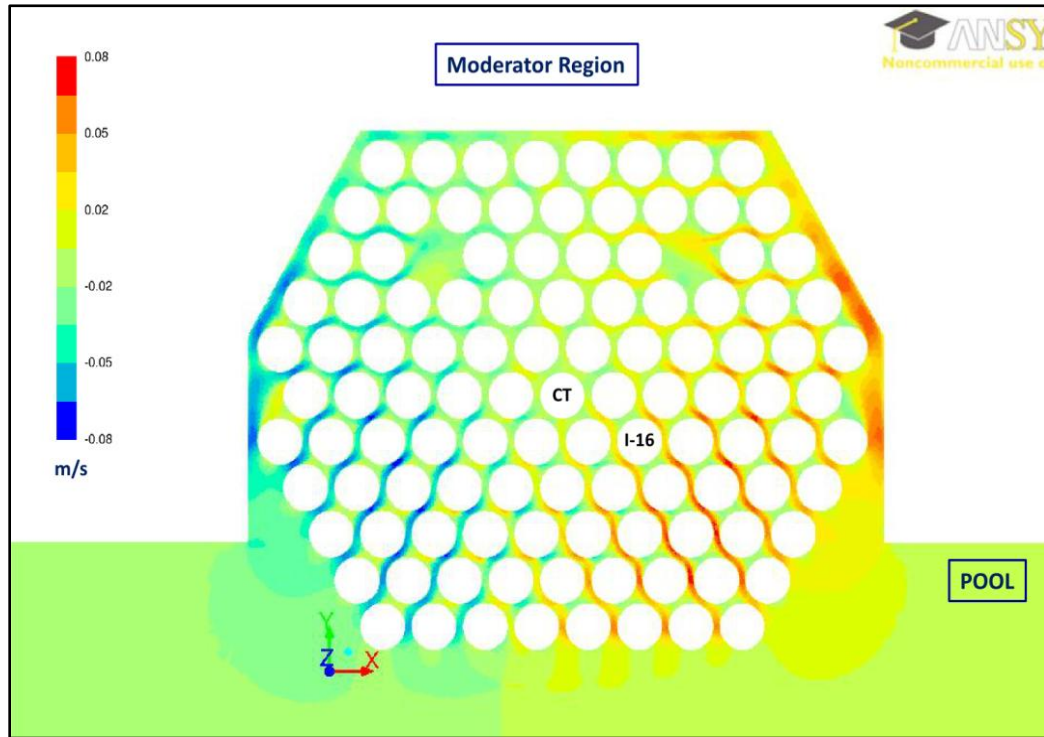


Figure 6-15: Radial velocity profile at the center of the core

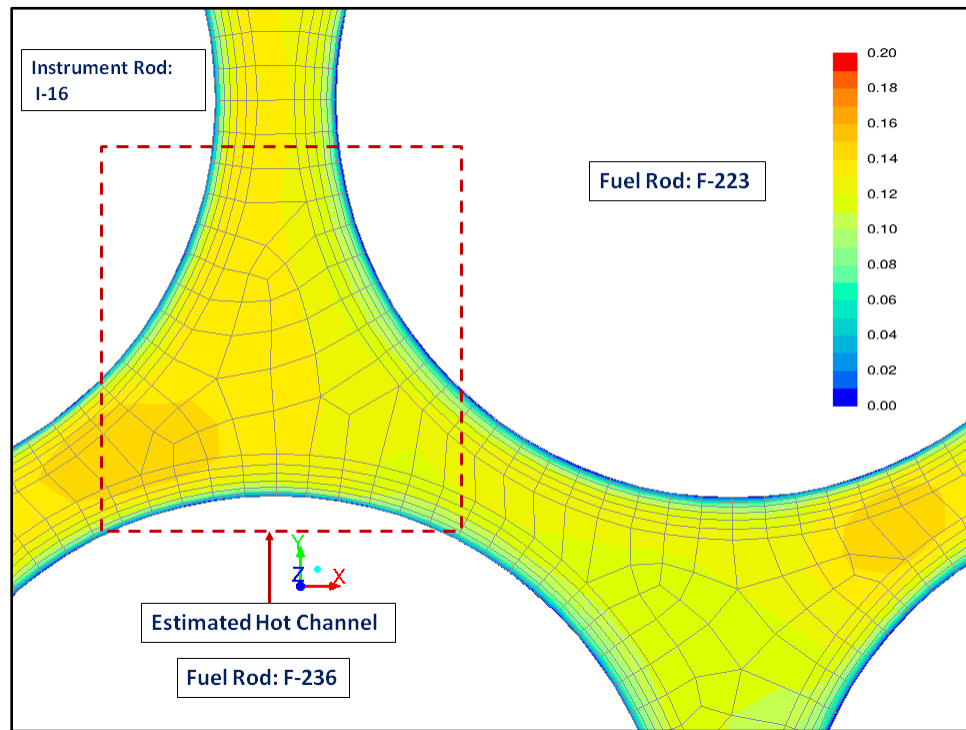


Figure 6-16: Contours of bulk fluid velocity (m/s) at the center of core and in the hot

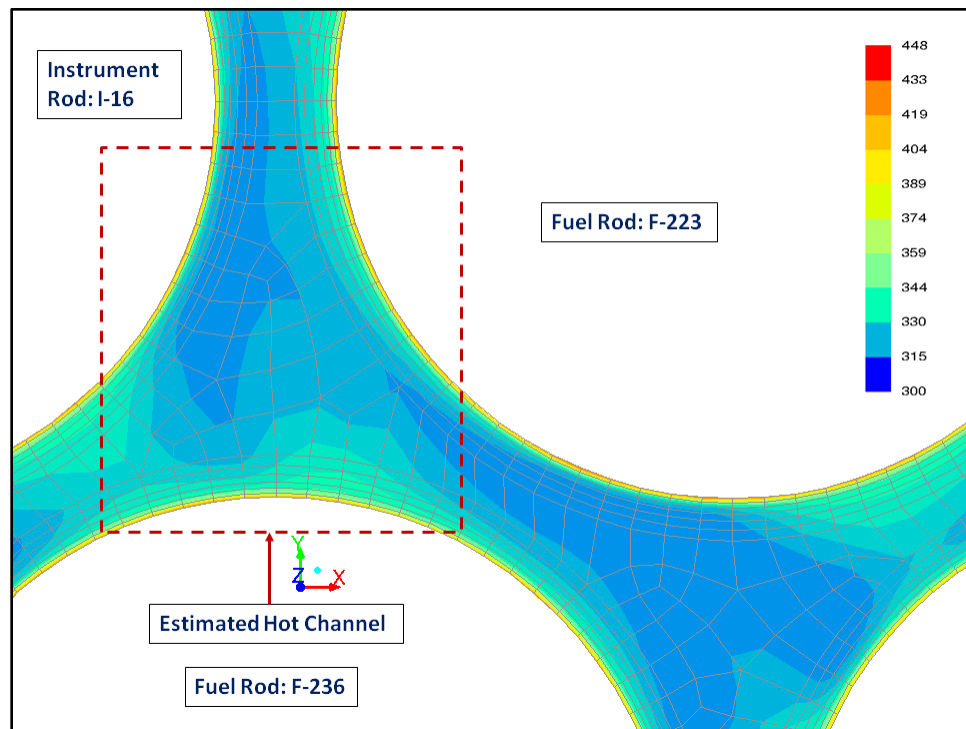


Figure 6-17: Contours of temperature (Kelvin) at the center of core and in the hot channel

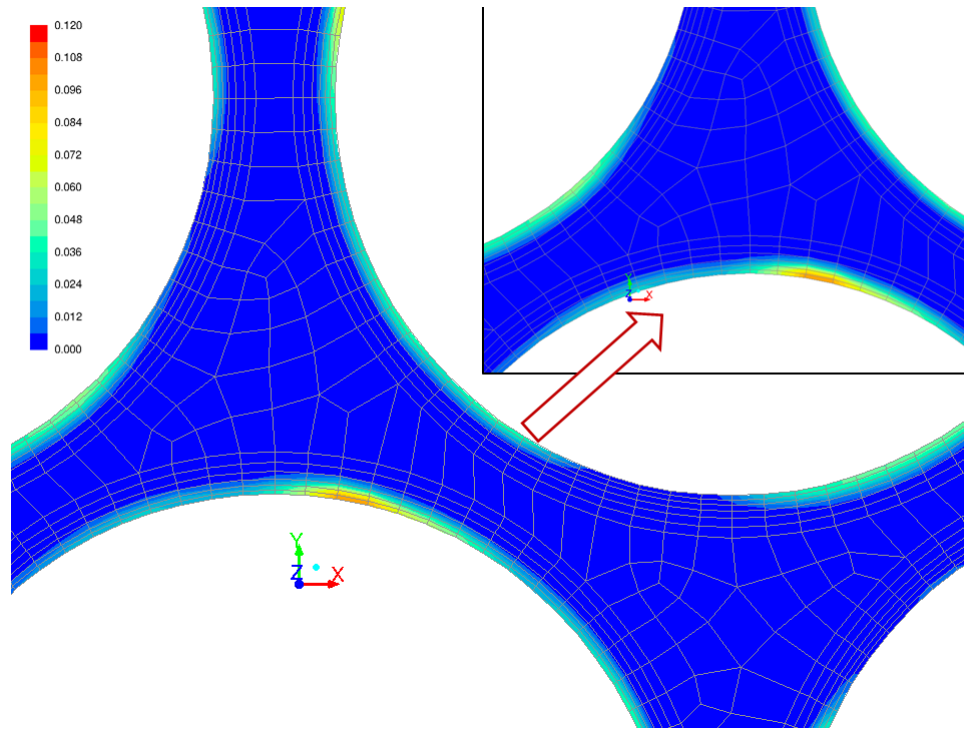


Figure 6-18: Contours of void fraction at the center of core and in hot channel

The axial temperature profile of the bulk fluid in Channel-R is given in Figure 6-19. The dashed lines in this figure indicate the active length of the fuel rods. The bulk coolant temperature at the bottom grid plate is 25 °C. The temperature of the coolant at the entrance of the Channel-R is 32 °C and at the channel exit is 88 °C.

Table 6-1 gives a comparison between measured and the CFD calculated bulk fluid temperatures at two measurement locations in Channel-R. The results clearly indicate a significant temperature increase in the new core-moderator assembly. This can be rationalized by a couple of issues stemming from both experiments and the CFD analyses. In the experiments, the thermocouple is driven by aluminum tube. The presence of the aluminum tube decreases the cross sectional area of the coolant channel. Under steady state conditions, the heat transferred to the fluid is constant while the flow area is not. Therefore, velocity of the coolant at that channel should increase by improving the heat transfer conditions. Another reason is due to the moderator shape. The new design restricts the cross-flow from pool to core. Therefore, the bulk fluid

temperature increases due to the flow redistribution. Although the temperature rise is not negligible, the bulk fluid temperature is estimated below the saturation temperature (111.4 °C) of the fluid at operating pressure of 1.5 atm. In the following section, the fuel temperature in the I-16 was determined in order to verify the safety of the new core-moderator assembly design for the PSBR.

In addition to bulk fluid temperature measurements in the existing PSBR, another comparison was achieved by using the experimental flow temperature profiles measured by Haag (1971) [23]. Figure 6-20 shows the bulk fluid temperature profiles along the active fuel length in four flow channels (see Figure 6-8) between the measurements and the CFD simulation results. The major difference between these results is the temperature decrease above the fuel center observed in the measurements. As mentioned in Chapter 2, the cross-flow entering into the core is the main cause of this temperature decrement above the core center in the existing PSBR design. In the new PSBR design, crescent moderator tank significantly reduces the cross-flow. Therefore, a steady increase in the bulk fluid temperature is observed along the flow channels.

The Fluent CFD results are also consistent with the Gougar's experiment [25]. As mentioned in the literature review for the thermal-hydraulics analysis of the PSBR, Gougar installed shrouds around the core to restrict the cross-flow entering into the PSBR core and then measured the temperature change in the fuel rods. A 3-°C temperature increase in the fuel rods above the core center was observed under this condition. In the new core design, the increase in the bulk fluid temperature in the flow channels was higher above the core center similar to Gougar's observation. However, the coupling of the core with the crescent moderator made this temperature rise significant as seen from Figures 6-19 and 6-20.

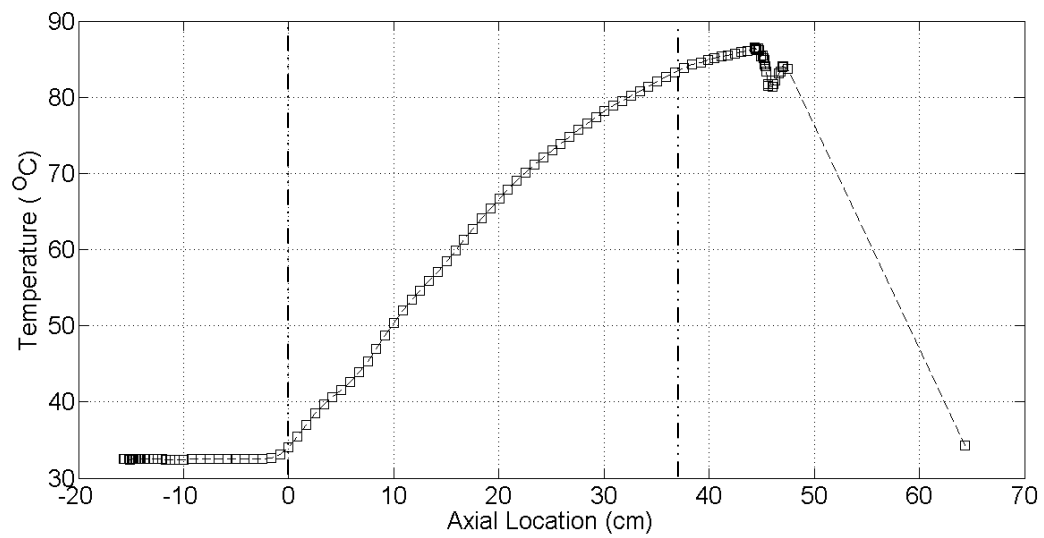


Figure 6-19: The axial bulk fluid temperature profile in Channel-R

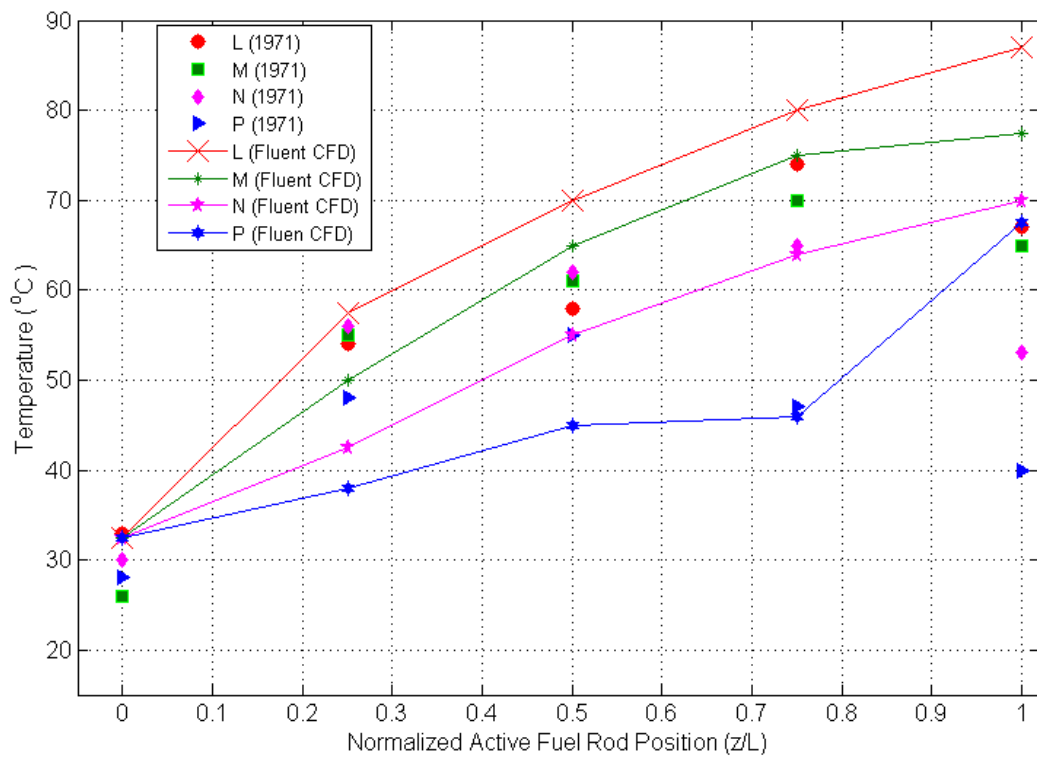


Figure 6-20: Comparison of the coolant temperature profiles in four flow channels (L, M, N and P) between experimental results (Haag, 1971) and Fluent CFD simulation results

Table 6-1: Comparison of CFD calculated axial coolant temperatures in Channel-R with the Cobra-TF (CTF) simulation results [28] and measurement in the existing PSBR design (2009) at two locations: (1) core center and (2) 12.7 cm above the core center

Location (cm)	Existing Reactor Design		New Reactor Design
	Measured Temperature (°C)	Cobra-TF Calculated Coolant Temperature (°C)	Fluent -CFD Calculated Coolant Temperature (°C)
19.05 (core center)	58.00	54.00	65.42
31.12	59.30	66.00	78.88

Calculation of the Fuel Temperature in I-16:

The fuel temperature (T_{fuel}) of I-16 was determined using the CFD calculated clad outer temperature ($T_c = 174.85$ °C) in following formula [60]:

$$T_{fuel} - T_c = \frac{q'''r_f}{2} \left[\frac{r_f}{2k_f} + \frac{1}{h_g} + \frac{t_c}{k_c} \right] \quad (6-18)$$

where q''' is the volumetric heat generation rate in unit of W/m^3 , r_f is the fuel radius, k_f is the thermal conductivity of the fuel, h_g is the gap heat transfer coefficient, t_c is the clad thickness, and k_c is the thermal conductivity of the clad.

Volumetric heat generation rate in the I-16 was predicted by TRIGSIMS code as $5.33 \times 10^7 W/m^3$. For the gap heat transfer coefficient, the correlation given by General Atomics was employed [74].

$$h_g = 0.0239q^2 - 1.4372q + 1593.1 \quad (6-19)$$

where q is the power in kW. Then, h_g was determined as $9228 W/m^2/K$ at 1 MW power. For TRIGA fuel, thermal conductivity of the fuel and clad are suggested as $18 W/m/K$ and $14 W/m/K$, respectively [40]. The clad thickness was calculated by assuming that the gap thickness was 0.0025 cm in the fuel rods [28].

Fuel temperature in the I-16 was found as 482 °C as a result of this calculation. On contrary, the measured fuel temperature is 540 °C in the same rod. The difference between these two results is mainly caused by the gap heat transfer coefficient and thermal conductivity of fuel employed in the calculations. The data provided by General Atomics are just recommended values and only gives an estimate of the actual values. General Atomics recommends three hypotheses which differ by ~50%. However, gap heat transfer coefficient and fuel thermal conductivity significantly changes as a function burnup during the lifetime of the fuel rods. Fuel thermal conductivity decreases as the burnup of the fuel rod increases.

The I-16 is first loaded to core in 1999 and it is still the hottest rod in the PSBR core. Therefore, it is expected a significant increase in the burning of the fuel rod and in the fission gas buildup in the gap region. This implies that fuel volume is less than the original loading and the gap region is thicker. It is hard to predict whether the fission gas buildup in the gap region will either degrade or improve the heat transfer coefficient. Since the I-16 is still providing the same amount of power to the core, volumetric heat generation rate used in Equation (6-18) should be higher since the volume in the fuel region is smaller. This means that estimated fuel temperature should be considerably higher than the calculated value. However, it is not expected to see a temperature increase in the fuel region which will exceed the safety limit of 1150 °C under steady-state operating conditions. Therefore, both the maximum fuel temperature and the maximum cladding outer surface temperature in the new core-moderator assembly do not compromise the safety limits.

Chapter 7

NEUTRONIC ANALYSIS of the NEW PSBR DESIGN with FIVE NEW BEAM PORTS

The final analysis that needs covered for the safety and optimum performance of the proposed PSBR is the neutronic analysis of the new PSBR core-moderator assembly with five new beam port designs. In Chapter 5, the neutronic design of the new core-moderator assembly was performed with five beam port models which were kept identical to BP4 in the existing facility. However, the design features of each neutron beam port will be different based on the experimental facility in which the beam port will be utilized. The list of the experimental facilities and neutron beam techniques that will be added to the existing beam port facilities was previously introduced in Table 1-1. Five additional experiments which will be implemented with the new design are: Triple-Axis Spectrometer (TAS) for scattering measurements, Prompt Gamma Activation Analysis (PGAA) for analysis of Boron and Hydrogen in technologically important materials, Neutron Depth Profiling (NDP) and Time-of-Flight (TOF)-NDP for near surface depth profile analysis of specific samples, and Neutron Powder Diffraction (NPD) for material characterization.

A preliminary study was performed before and given in M.Sc. thesis of Butler [21] for the instrument and source selection for the new PSBR facility. Based on this study, it is considered to utilize a cold neutron source for PGAA, NDP and NPD. Therefore, there will be at least three neutron guide tubes in the cold neutron beam port. All the other beam ports will utilize thermal neutron beam to the beam port facilities.

The name convention used for the thermal beam ports in Chapter 5 were followed here as such new beam ports were abbreviated as NBP1, NBP2, NBP3 and NBP4. Guide tubes were named as GT1, GT2 and GT3.

In the new PSBR neutron beam facility, the Triple-Axis Spectrometer (TAS) will be in NBP1. The neutron transmission and neutron imaging facilities which are functional in the existing facility will be performed in NBP3 and NBP4, respectively. The detailed information about each neutron beam port design was given in Section 7-1.

The neutron flux requirements were also determined in reference [21] by analyzing the lowest utilized thermal or cold neutronic flux for each instrument at other research reactors. These values are also tabulated in Table 7-1.

Table 7-1: Summary of instruments in the new facility and lowest utilized neutron flux to similar instruments at other research reactors [21]

Location	Instrument	Lowest Utilized Flux (neutrons/cm²/sec)
NBP1	Triple-Axis Spectrometer	1.72x10 ⁵ (thermal) 2x10 ⁵ (cold)
NBP2	Free Beam	1x10 ⁶ (thermal)
NBP3	Neutron Transmission	1x10 ⁴ (thermal)
NBP4	Neutron Imaging	3x10 ⁶ (thermal)
GT1	Prompt Gamma Activation Analysis	2.4x10 ⁷ (thermal) 5x10 ⁷ (cold)
GT2	Neutron Powder Diffraction	1x10 ⁵ (thermal)
GT3	Neutron Depth Profiling	1.5x10 ⁸ (thermal) 2.5x10 ⁹ (cold)

New Beam Port Designs

In this section, detailed information about the design features including the physical dimensions, filter material and collimator system of each new beam port are given with technical drawings.

NBP1 and NBP2

When designing NBP1 and NBP2, neutron beam port and collimators designed by Dr. Ünlü at the University of Texas (UT) research reactor was used [75]. The total length of this beam port in the UT facility is 318.6 cm and the beam port consists of two tubes made of steel. The first steel tube is 165.6 cm long and 15.4 cm in diameter. A single crystal sapphire filter, which is 17.27 cm long and 3.56 cm in diameter, is located at the end of first tube. On the other hand, the second steel tube is 160.3 cm long and 20.6 cm in diameter. The collimator system is installed into the second tube. Collimation is achieved by several steel and lead sections with some sheets of boral between the steel sections. This collimator design provides a high quality thermal neutron beam to the NDP facility of the UT reactor. The thermal neutron flux is 1.5×10^8 n/cm²-s at the exit of the collimator [75]. The diameter of beam size is 6.5 cm at the collimator entrance and 1.5 cm at the collimator exit.

In the new design, a similar beam port design to the NDP beam port in the UT facility will be implemented for NBP1 and NBP2. The dimensions of sapphire filter will be 7.6 cm (3 inches) in length and 4.4 cm in radius. These beam ports will consist of two steel tubes. The sapphire filter will be installed to end of the first tube and second tube will have the same collimator system to the NDP beam port. The length of the first steel tube will be 91 cm for NBP1 which is located at the tangent of the core with a relative tilt angle of $\sim 50^\circ$. On the other hand, the length of the first tube will be 142 cm for NBP2. The total length of NBP2 will be 300 cm. The second steel tubes in these beam ports will be located in the pool wall and will be identical to each other. A drawing of design features of NBP1 and NBP2 is shown in Figure 7-1.

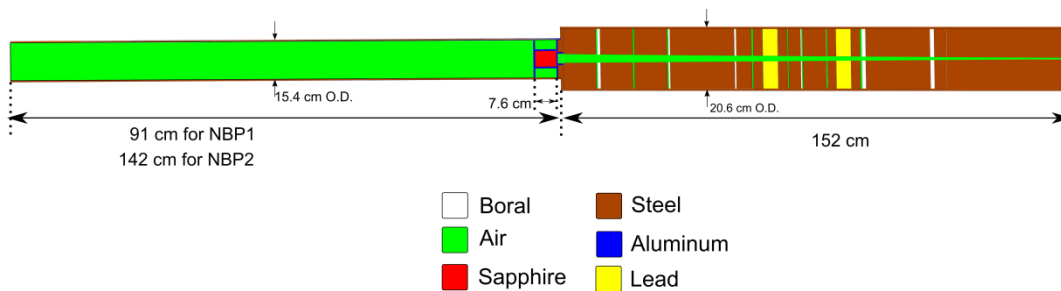


Figure 7-1: Design features of NBP1 and NBP2 in new PSBR facility

PSBR Cold Neutron Beam Port

Similar to NBP1 and NBP2, the PSBR cold neutron beam port will be identical to Texas Cold Neutron Source (TCNS) [76][77] with minor changes. A technical drawing of the TCNS is given in Figure 7-2. TCNS as a radial beam port of University of Texas (UT) research reactor is composed of an aluminum moderator chamber, a cooled mesitylene moderator, a cold source cryostat system and a neutron guide system. Mesitylene is a 1,3,5-trimethyl benzene which freezes at 228 K and boils at 437 K. The moderator chamber is a 7.6-cm diameter and 2-cm thick cylinder located in the graphite reflector surrounding the UT reactor core. A lead shield is placed in front of the vacuum chamber. The cryostat system consists of a Two-Phase Closed Thermosyphon with a Reservoir (TPCTR), a helium cryorefrigerator, and a vacuum box. Neon is used as the working fluid or the reservoir of the thermosyphon. It evaporates in the moderator and liquefies at the cold head of helium cryorefrigerator which is able to remove 9-W heat load at 20 K. The thermosyphon maintains the mesitylene moderator at below 30 K. Vacuum box provides insulation for the low temperature components. The pressure inside the vacuum box is 1×10^{-6} torr. The mesitylene moderator flows in a closed-loop system to reduce radiation damage to the moderator [76].

The cold neutrons in the beam port are extracted from two horizontal neutron channels. A 2-m long curved neutron guide section is located in one of the neutron channels and a 2-m long

section is followed by the neutron guide at the outside of the beam port. All the neutron guide sections provide 300-m radius of curvature and contain three vertical channels. The total length of the guide tube is 6-m and the guide tube walls are coated with ^{58}Ni layer with 1000-Angstrom thickness. An 80-cm converging guided neutron focusing system is located at the end of neutron guide. The neutron focusing system is coated with nickel carbon titanium supermirrors. The main function of the neutron focusing system is to intensify the neutron beam for neutron capture experiments. The converging guide is composed of four truncated rectangular cone elements and each element is made of 20-cm long single-crystal silicon plates. The characteristics energy of the neutron guide is 11 meV [76] in the TCNS.

The PSBR cold neutron beam port will have three steel tubes. The first tube will be 20 cm in diameter and 40 cm in length. This section will also have 5-cm thick lead shield on the front surface. The moderator chamber will be installed to the first tube at a distance which is 15 cm away from the lead shield. The second tube will be 25 cm in diameter and 111 cm in length. Three neutron guide tubes will be installed to the beginning of the second tube and their length will be about 15 meters with a curvature of a 300-meters radius. The third steel tube will be 30 cm in diameter and 148 cm in length. All tube sections will be kept under vacuum. Guide tubes will be identical to each other and located 1 cm away from each other. They will be installed at the same elevation and directed to the moderator chamber center in order to obtain the maximum cold neutron flux. The coating material on the guide tubes will be a supermirror that has not been selected yet.

The geometric and thermo-physical factors affecting the TPCTR cooling system utilized in the TCNS was previously investigated by Dr. Habte with a detailed CFD analysis [78]. The TCNS is not capable to accommodate 4-W heat loads with a maximum moderator temperature of 44 K. A better cooling system which increased the operating range and heat removal capacity of the TPCTR was designed for the PSBR cold neutron source. In light of the calculated design

parameters, new TPCTR cooling system will benefit a heat load of about $\sim 10\text{W}$ with operating temperature of 28 K. Some of the design changes recommended for the TCNS are summarized as material selection for the thermosyphon, operating pressure and size of evaporator section, and size of gas reservoir and the thermosyphon. As a result of this study, copper or aluminum and neon gas are suggested as the construction material and working fluid of the thermosyphon, respectively. The other design parameters calculated in the optimum design was discussed and given in Ph.D. thesis of Dr. Habte [78]. The temperature of the mesitylene in the ideal design will be $\sim 26\text{ K}$ and this new TPCTR will be utilized in the new PSBR cold neutron source.

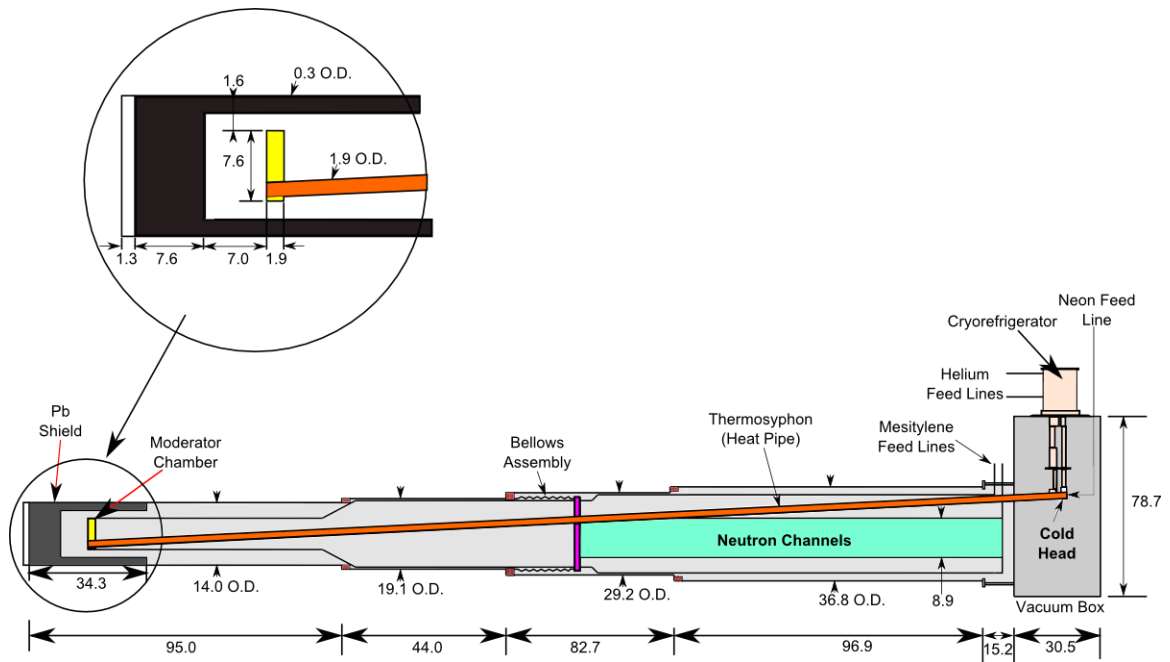


Figure 7-2: A schematic layout of the University of Texas Cold Neutron Source (UTCNS) (dimensions are given in cm)

NBP3

NBP3 is identical to design of BP7 in the existing facility. Figure 7-3 shows the existing BP7 which consists of four sections which are connected by six different parts. The length of the first three sections is 29.5 inches and the last section is 28-inches long. These sections have 5%

borated poly pellets, sandcrete, and lead in order. Figure 7-4 shows the cross sectional views of each part. There are three holes in each part except in Part-C. Part-A and Part-F consist of 2-inches thick and 5% borated polyethylene sheet. On the other hand, Part-B, Part-C and outer support are made of 0.5-inches thick aluminum plates. Part-D and Part-E are made of 1-inch thick lead.

The support frame and graphite interface box used in the existing BP7 design will not be utilized in NBP3. Beside that all the design characteristics will be identical between the existing and new transmission beam ports.

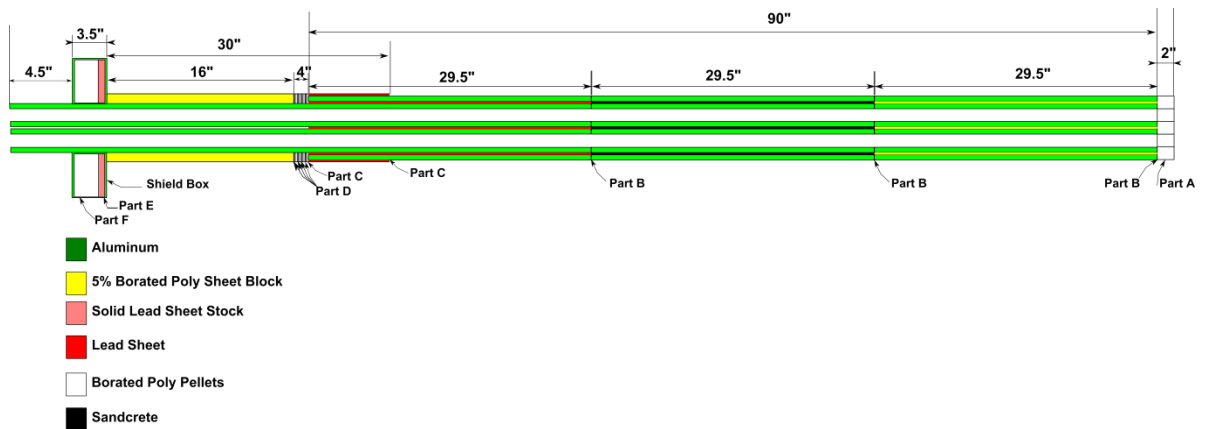


Figure 7-3: BP7 design in the existing reactor design

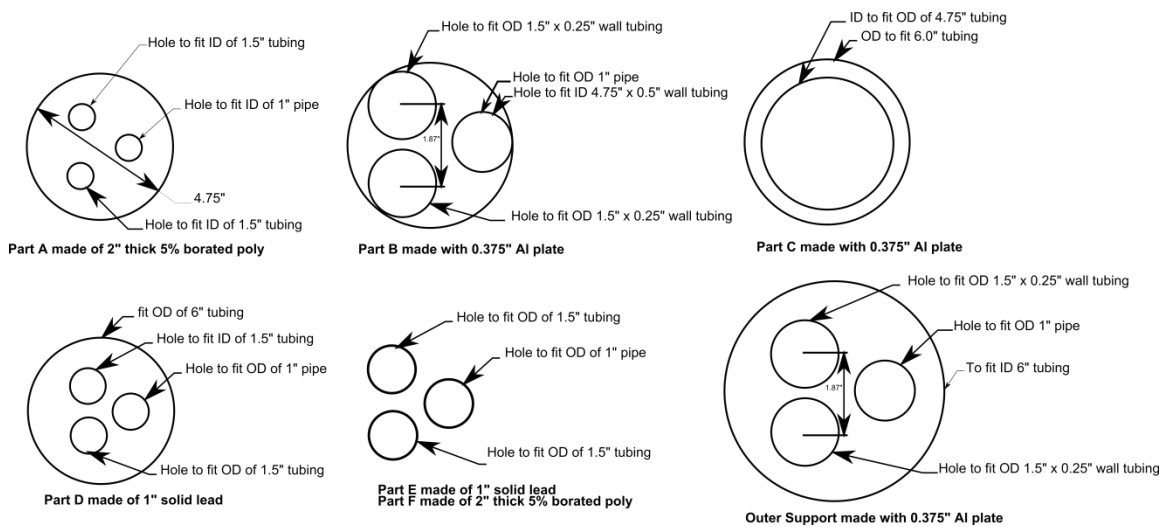


Figure 7-4: Cross sectional view of each section in BP7

NBP4

NBP4 will be same with existing BP4 which was previously given in Figure 3-9. There will be a collimation section in which bismuth crystal, lead and boral are installed in order. The collimation section will be close to crescent moderator. The collimator will be installed into the last section similar to existing BP4.

The Neutronic Modeling of Proposed New Beam Ports using MCNP Code

The neutronic analysis of the final design with the proposed beam port designs was achieved by using MCNP5 code. Core loading 53H at its initial load as of May, 2009 was the reference core model in the MCNP analyses. Fuel and control rods in this loading pattern were modeled with their depleted compositions calculated by the Burned Coupled MCNP Simulation tool developed for the PSBR core [79]. The Burned Coupled MCNP Simulation tool was developed upon the MURE code with custom C++ object implementations. It provides very accurate and up-to-date burnup history of TRIGA fuel starting from 1965. A new cross section data set generated using NJOY-99.264 code from ENDF/B-VII data tapes [29] for 86 isotopes with 10 °C temperature increments between 293.72 K and 900 K was implemented to this simulation tool. On the other hand, TRIGSIMS code employs a cross section data set generated for 20 selected important isotopes with 50 °C temperature intervals. Accurate modeling of fuel temperatures in the PSBR core is very important because the TRIGA fuel is significantly affected by the temperature changes. As the fuel temperature increases, the probability that an incoming neutron can gain energy after a scattering event by the hydrogen in the TRIGA fuel also increases. This process is called up-scattering and adversely changes the neutron economy in the core. The new cross-section data set generated with 10 °C temperature intervals is expected to

accurately model this affect. Additionally, it was verified that the Burned Coupled MCNP Simulation tool with an improved cross section data set provides very accurate and up-to-date burnup history of TRIGA fuel compared to TRIGSIMS [79].

In order to take advantage of the new cross section data set, the fuel temperatures in each fuel and control rods were calculated and defined with temperature (TMP) cards in the MCNP models. The fuel temperatures were estimated by making a linear interpolation between the measured fuel temperatures in instrument rod (I-16) and the Burned Coupled MCNP Simulation tool calculated local power generation in each fuel and control rod.

MCNP model of the new PSBR design with new beam ports (with three guide tubes) are shown in Figure 7-5. MCNP calculations were also performed in two-steps defined by full core and beam port models with conical surface source direction biasing technique and point detector estimators. In the full core model, the number of active cycles was 500 and 1 million particles per cycle were simulated. The simulations were completed until the estimated relative error in the thermal neutron region became less than 0.05 (5%) for point detectors (F5-tally) in each beam port model.

The neutron and gamma flux spectra calculated at the exit of each beam port were given in the following subsections. The scaling factor calculated in Equation (5-6) was used to convert the normalized MCNP tally results for the neutron flux to the 1 MW power operation results. The scaling factor estimated for the gamma flux and given in Equation (5-11) was not applicable to this analysis since it was calculated for the existing BP4 model. However, new beam port designs in this case are different and the scaling factor will be different for each beam port. Therefore, gamma spectra calculated by the MCNP code per total numbers of starting source particles were employed in the neutronic performance evaluation of the new beam ports.

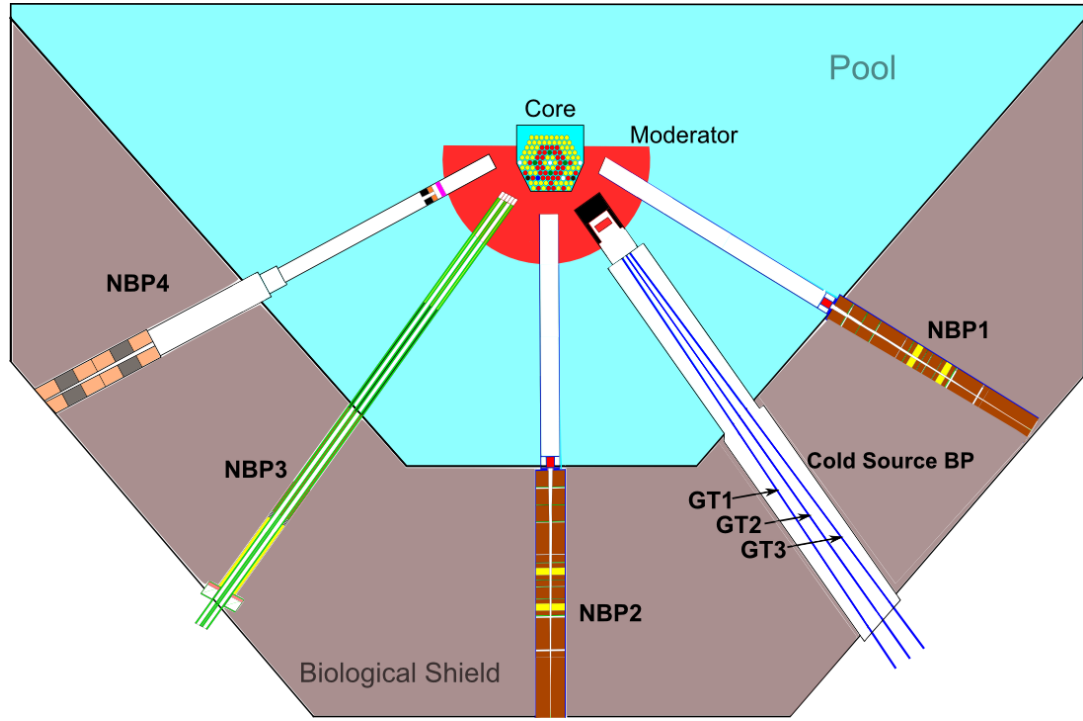


Figure 7-5: A schematic layout of the final PSBR design with four thermal neutron beam ports and one cold neutron beam port in which three guide tubes are channeled

Modeling of NBP1 and NBP2

The MCNP calculation methodology for NBP1 and NBP2 were different than the others because of the presence of sapphire filter. As mentioned in Chapter 2, there is no available cross section data for sapphire filter in evaluated neutron data tables and an experimental data measured at NIST had been previously utilized to estimate the neutron transmission rate in the hemlite-grade sapphire crystal. A 10% difference was observed between the measurements and simulation results with this approach. In this case, a neutron attenuation data obtained with time-of-flight measurements and fitted to Cassels function by Mildner [44] was employed. The Cassels function to define the attenuation cross section (in barn) for hemlite-grade sapphire filter was given as

$$\sigma(\lambda) = A\lambda + C[1 - (\lambda^2/2B)(1 - \exp\{-(2B/\lambda^2) + (D/\lambda^4)\})], \quad (7-1)$$

where λ is the neutron wavelength. The fitting parameters used in this equation were given as $A = 8.61 \mu m^{-1} nm^{-1}$, $C = 0.0276 mm^{-1}$, $2B = 5.84 \times 10^{-3} nm^2$, and $D = 2.35 \times 10^{-5} nm^4$.

A comparison between the measured neutron cross section for hemlite grade sapphire as a function of energy and Cassels formula was also achieved in reference [44]. Although some discrepancies at fast energy region were reported, the Cassels formula gave very accurate results in the thermal energy region. The attenuation cross section as a function of energy calculated for hemlite-grade sapphire filter is plotted in Figure 7-6. The conversion from wavelength to energy was attained by using the de Broglie equation [80]:

$$\lambda = h/\sqrt{2Em}, \quad (7-2)$$

where h stands for Planck's constant and m is the neutron mass.

The neutron transmission rate in sapphire filter was determined by using Cassels function and the neutron current spectrum calculated on the sapphire filter entrance surface by MCNP simulations. If the incoming neutron current on the filter entrance surface is defined as I_i , the transmitted neutron current from the filter, I_t can be determined by using the following formula.

$$I_t = I_i(E) \exp(-N_d \sigma(E)t), \quad (7-3)$$

where N_d is the number density of Al_2O_3 molecules and t is the thickness of the filter. The number density was reported as $2.35 \times 10^{22} cm^{-3}$ [44] and the thickness of the sapphire filter installed to the NBP1 and NBP2 is 7.62 cm (3 inches). The estimated thermal neutron transmission and fast neutron attenuation rates are given in Table 7-2. The relative errors in these results were determined using the MCNP estimated relative error for the incoming neutron current spectrum. The calculated thermal neutron transmission rate in hemlite sapphire filter closely matches with the measured value (74%) by Dr. Ünlü (1994). However, the measured (90%) and the MCNP calculated fast neutron attenuation rates in hemlite-grade sapphire filter are significantly different from each other. This is due to the reported discrepancy of the Cassels function in the fast neutron

energy region [44]. Beside that it was assumed that the total attenuation cross section will be constant in the fast energy region, which is actually not.

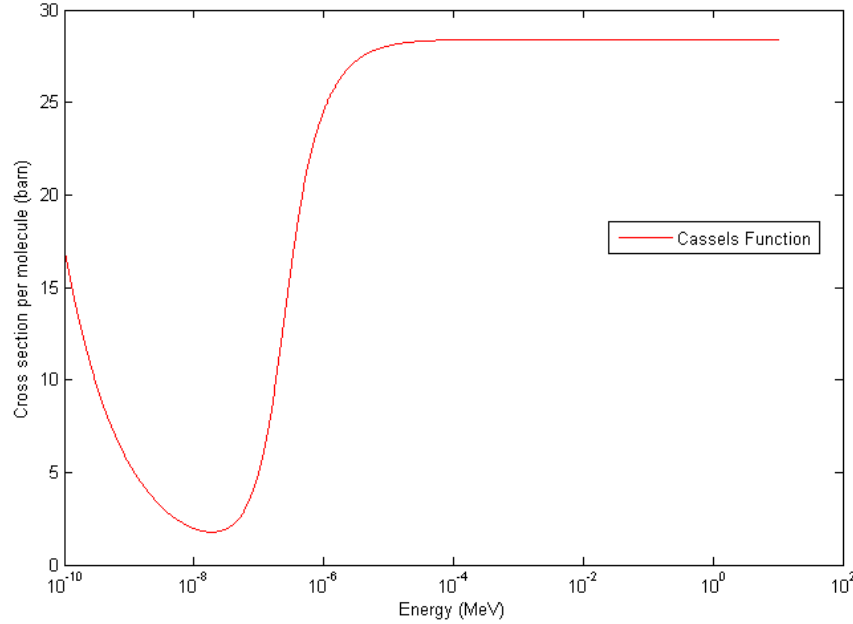


Figure 7-6: The calculated attenuation cross section as a function of energy in 7.6-cm thick hemlite-grade sapphire filter

Table 7-2: The estimated thermal neutron transmission and fast neutron attenuation rates in hemlite sapphire filter

Beam Port	Thermal Neutron Transmission Rate (%)	Fast Neutron Attenuation Rate (%)
BP1	74.91 ± 3.56	78.15 ± 5.60
BP2	73.32 ± 2.43	78.15 ± 5.60

The calculation methodology for NBP1 and NBP2 with hemlite sapphire filter can be summarized as follows. The MCNP calculations performed in two-stages with full core and beam port models were followed with an additional calculation in the sapphire filter. The angular neutron current spectrum was calculated on the entrance surface of the sapphire filter in the full core model for 1 MW power operation. Then, the transmitted neutron current spectrum from the sapphire filter was determined by applying Equation (7-3). Since the angular distribution of

neutrons on the filter entrance surface were assumed to be same after the transmission, the same source biasing parameters calculated at the entrance surface was applied to the transmitted neutron beam. Then, the neutron and gamma flux spectra were predicted at the exit of new beam port model by using the point detector estimators.

Modeling of PSBR Cold Neutron Beam Port

The length of the guide tubes in the PSBR cold neutron beam port will change based on the coating material selection because the reflection power and loss probability per reflection are different between candidate coating materials such as supermirror (NiT), Beryllium, BeO, Nickel, diamond, etc. ⁵⁸Ni coating material was used in the MCNP calculations as an initial guess. Beside that the guide tubes were modeled up to the biological shield exit although their lengths will be about 15 meters in the final design. Thus, the MCNP analysis will give the neutron and gamma spectra at the exit of the cold neutron beam port. MCNP calculations were performed in two steps with full core and guide tube models. The angular neutron and gamma current spectra were first calculated on the entrance surface of each guide tube in the full core model. Then, the neutron and gamma flux spectra at the exit of each guide tubes were calculated by defining the source biasing parameters on the source surfaces and the point detectors at the exit of guide tubes. The mesitylene moderator temperature was set to 20 K during the calculations.

The cold neutron flux is defined by the thermal-equivalent flux which is calculated as

$$\phi_0 = \frac{\sqrt{\pi}}{2} \frac{v_0}{v} \phi_R. \quad (7-4)$$

In this equation, ϕ_R is the real flux in the cold neutron energy region (between 0.1 meV and 10 meV), v_0 (2198 m s⁻¹) is the thermal neutron velocity and v is the velocity of the neutron at medium temperature.

Modeling of NBP3 and NBP4

The MCNP calculation approach performed in two-steps with conical source biasing and point detector estimators were also applied to the neutronic analysis of NBP3 and NBP4 as usual. In this case, surface source were fabricated on a hypothetical surface defined at 2-m deep inside of these beam port models. For NBP3, the neutron and gamma flux spectra were calculated inside the BF3 detector installed to the end of one of the tubes inside this beam port. For the NBP4, tally results were obtained at the exit of the collimator with 1.5 cm inner diameter.

Results and Discussion

The MCNP calculated neutron and gamma flux spectra at the end of each new beam port and guide tubes are given between Figures 7-7 and 7-20. Additionally, the total thermal and total fast neutron flux are given in Table 7-3. In this table, the tabulated total thermal neutron flux values for the guide tubes represent the total thermal-equivalent neutron flux. The total gamma flux values at the same beam port locations are given in Table 7-4.

NBP1

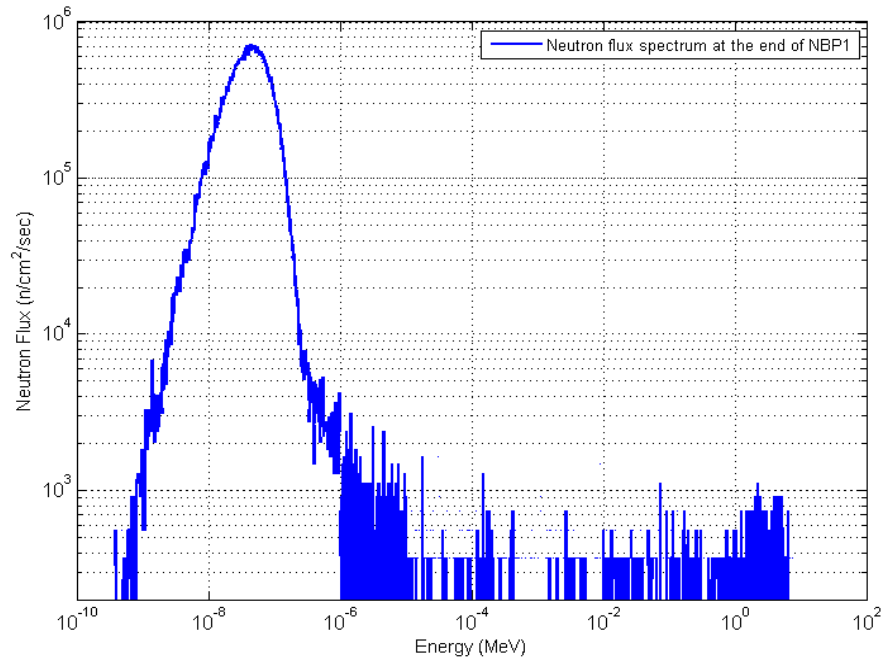


Figure 7-7: Neutron flux spectrum calculated at the exit of NBP1 in the final PSBR design

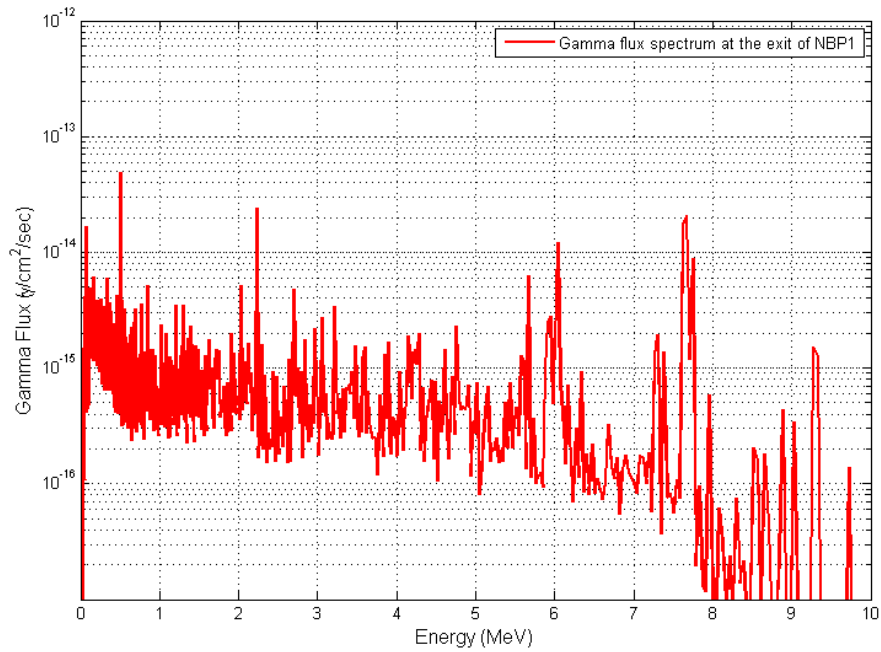


Figure 7-8: Gamma flux spectrum calculated at the exit of NBP1 in the final PSBR design

NBP2

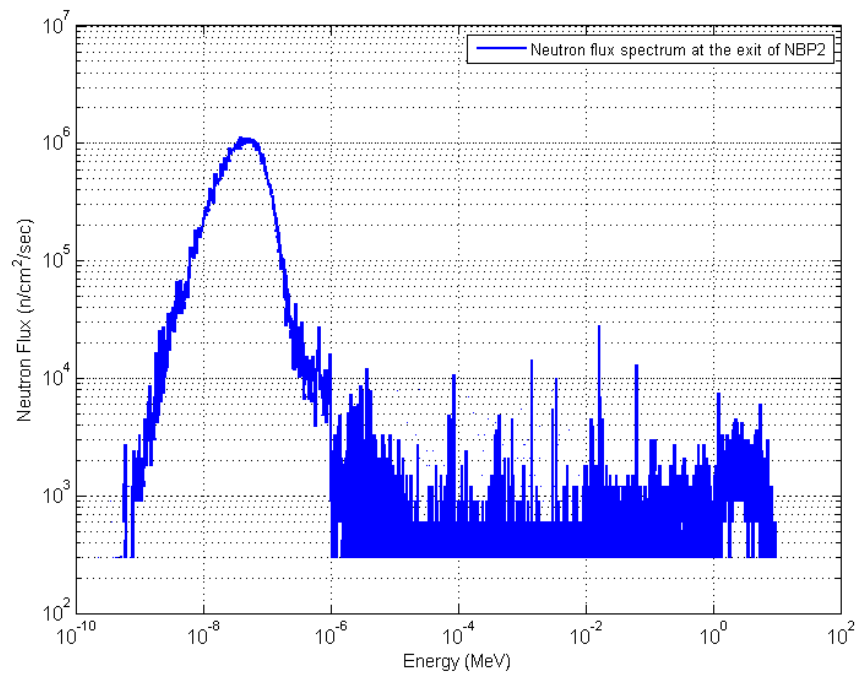


Figure 7-9: Neutron flux spectrum calculated at the exit of NBP2 in the final PSBR design

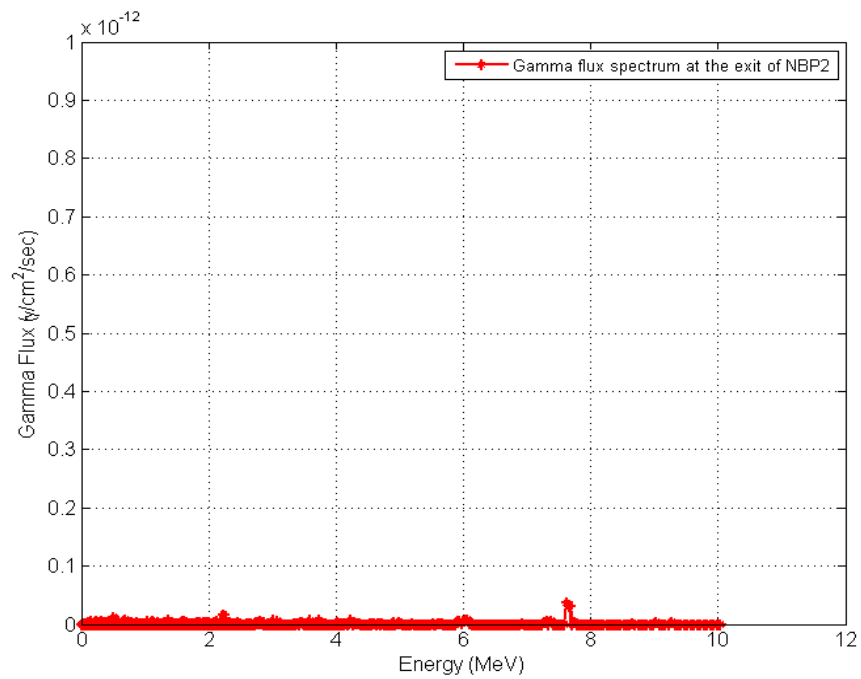


Figure 7-10: Gamma flux spectrum calculated at the exit of NBP2 in the final PSBR design

NBP3

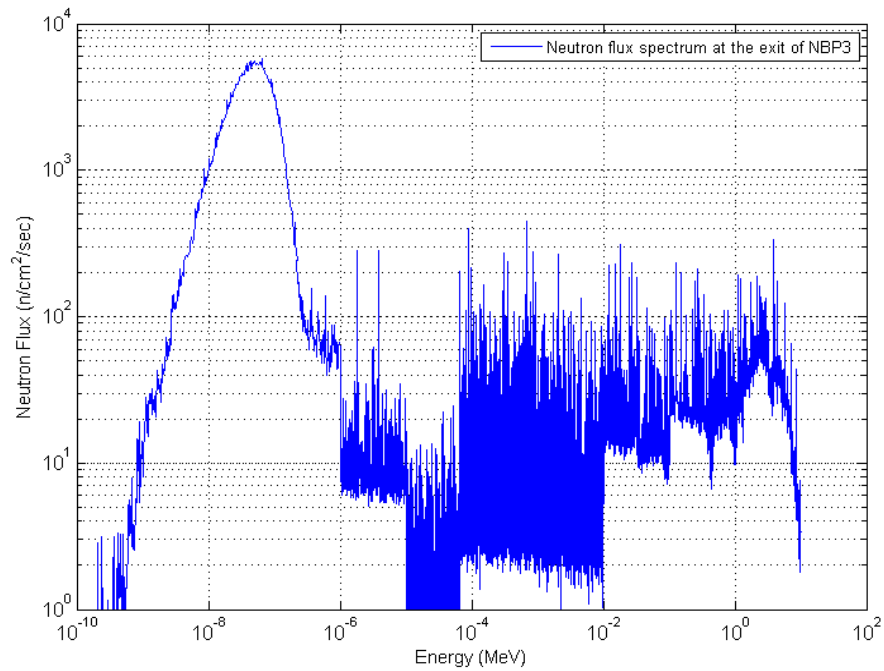


Figure 7-11: Neutron flux spectrum calculated at the exit of NBP3 in the final PSBR design

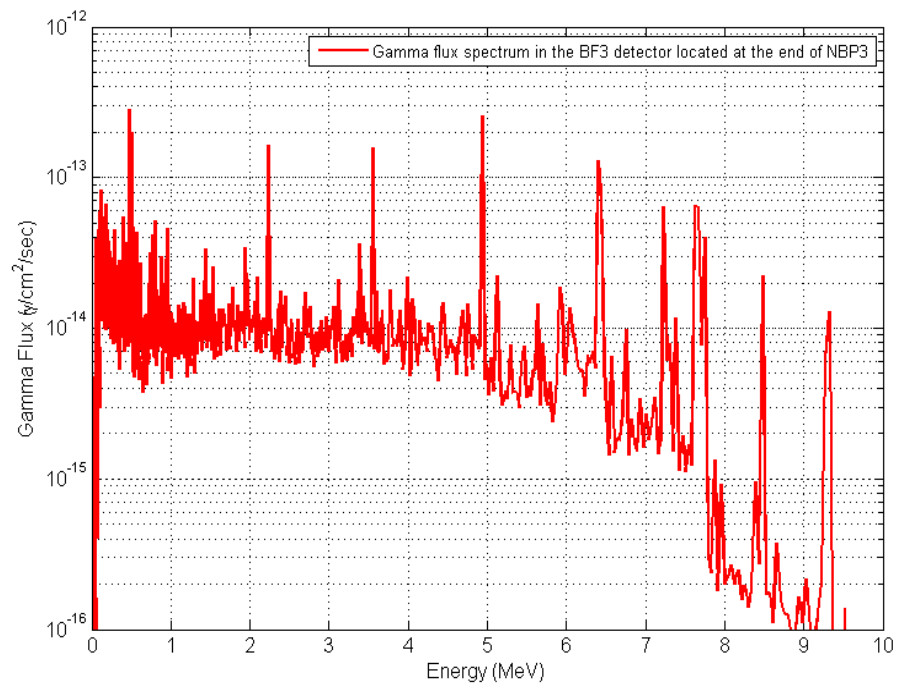


Figure 7-12: Gamma flux spectrum calculated at the exit of NBP3 in the final PSBR design

NBP4

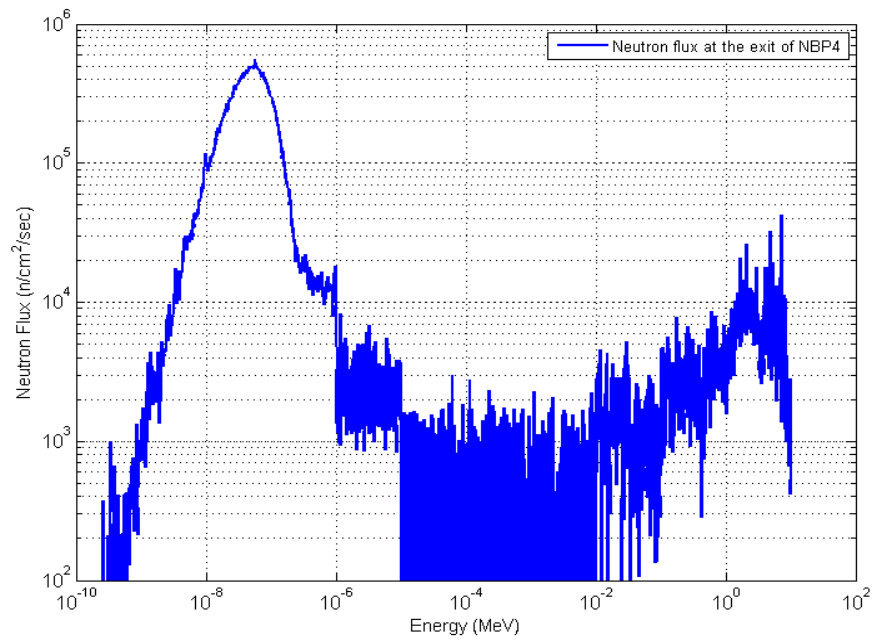


Figure 7-13: Neutron flux spectrum calculated at the exit of NBP4 in the final PSBR design

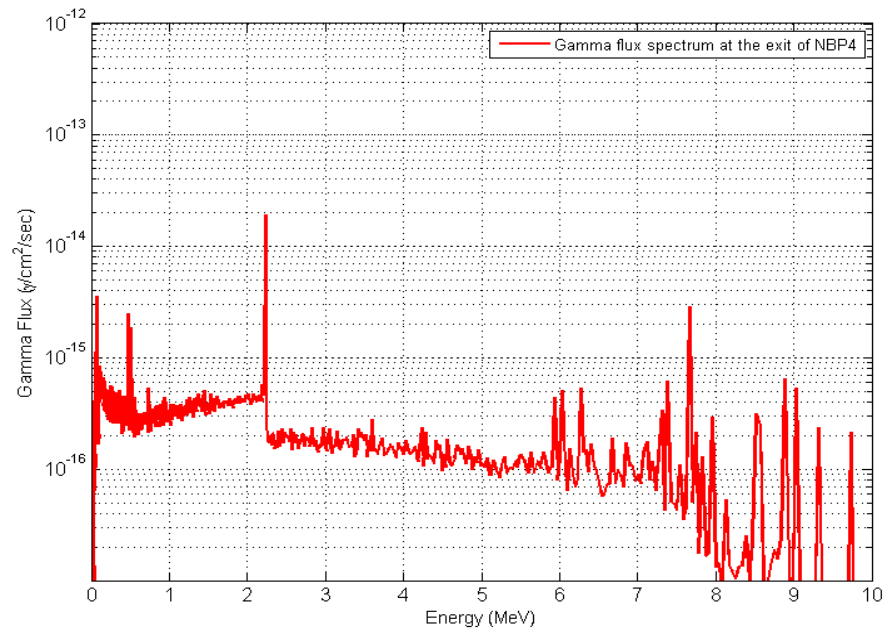


Figure 7-14: Gamma flux spectrum calculated at the exit of NBP4 in the final PSBR design

PSBR Cold Neutron Source

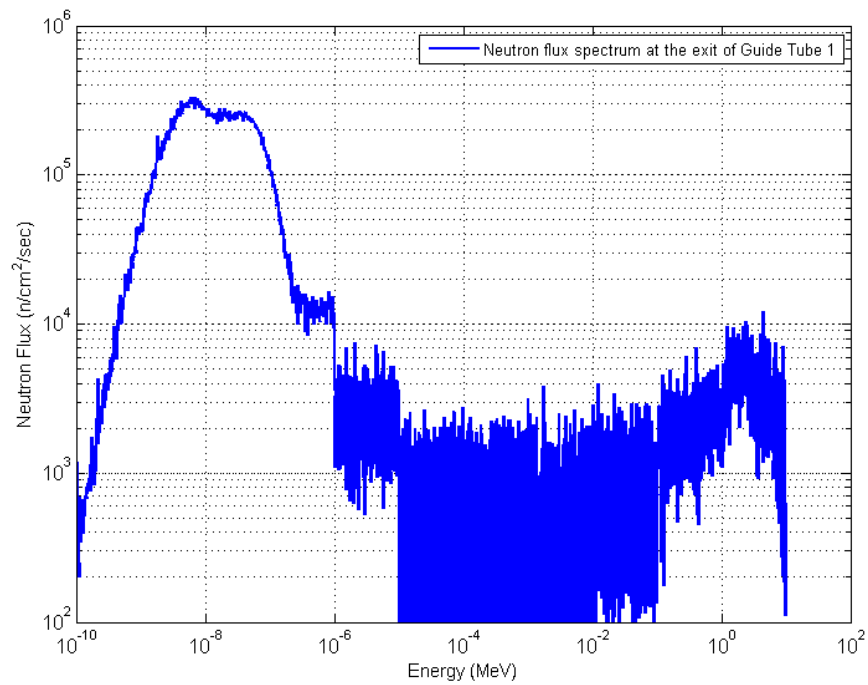


Figure 7-15: Neutron flux spectrum calculated at the exit of Guide Tube 1 in the final PSBR design

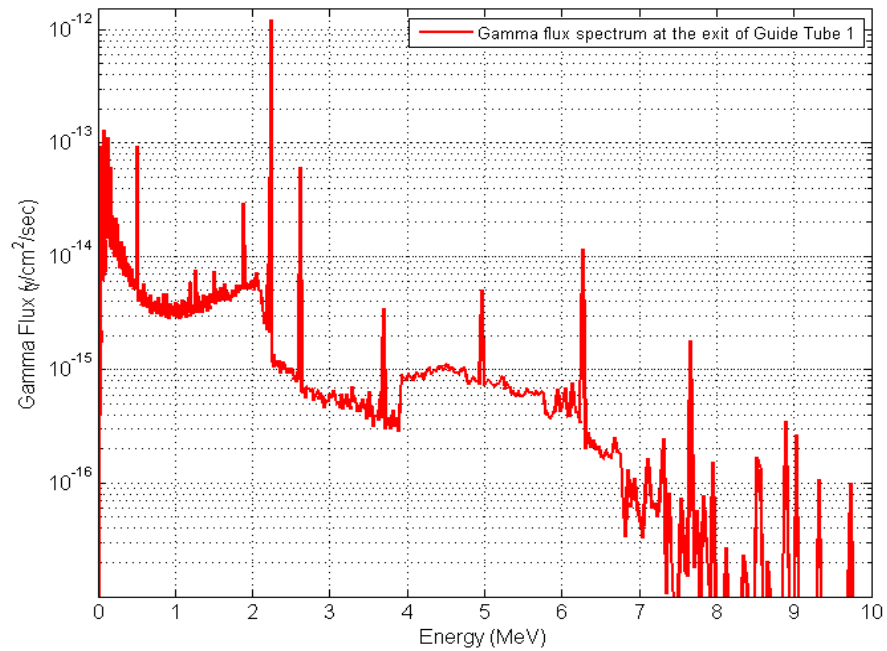


Figure 7-16: Gamma flux spectrum calculated at the exit of Guide Tube 1 in the final PSBR design

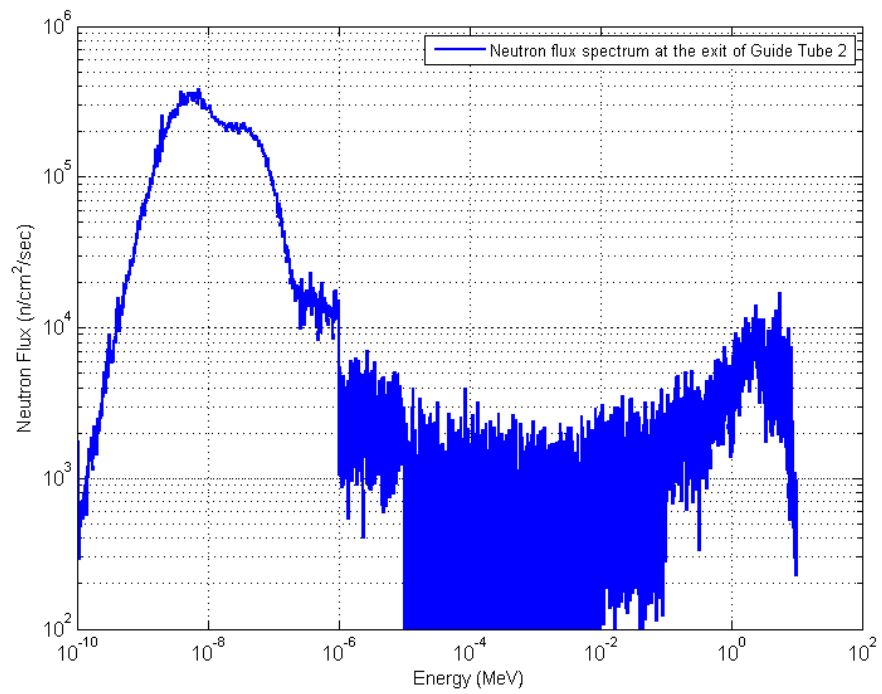


Figure 7-17: Neutron flux spectrum calculated at the exit of Guide Tube 2 in the final PSBR design

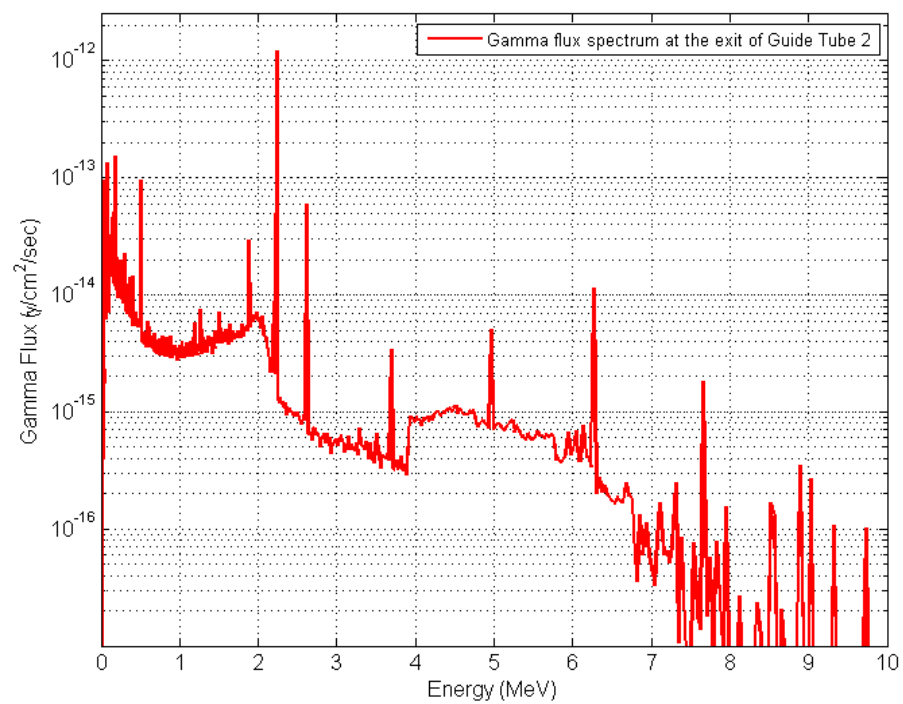


Figure 7-18: Gamma flux spectrum calculated at the exit of Guide Tube 2 in the final PSBR design

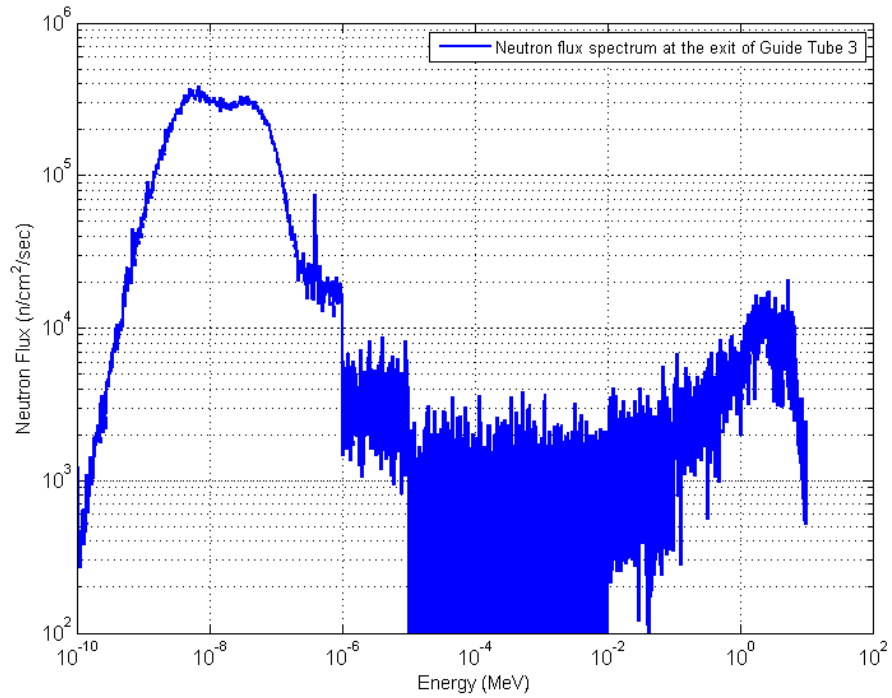


Figure 7-19: Neutron flux spectrum calculated at the exit of Guide Tube 3 in the final PSBR design

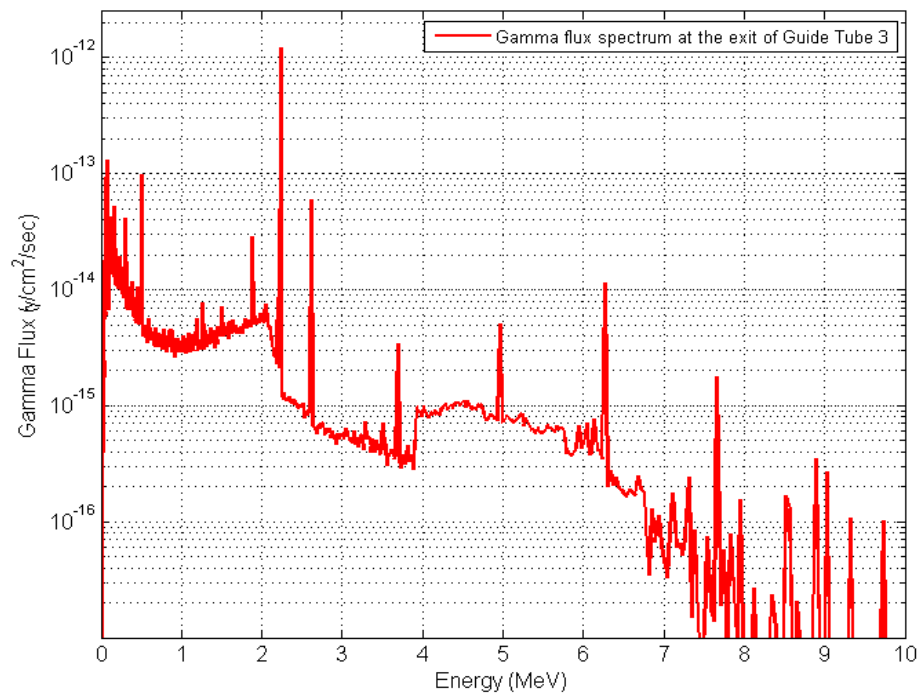


Figure 7-20: Gamma flux spectrum calculated at the exit of Guide Tube 3 in the final PSBR design

Table 7-3: Total thermal and total fast neutron flux calculated at the exit of new beam ports and guide tubes in the final PSBR design

Beam Port	Total Thermal Neutron Flux (n/cm²/sec)	Error (%)	Total Fast Neutron Flux (n/cm²/sec)	Error (%)
NBP1	5.43x10 ⁷	3.28	1.41x10 ⁵	5.41
NBP2	8.67x10 ⁷	4.80	9.19x10 ⁵	7.46
NBP3	4.39x10 ⁵	3.80	2.09x10 ⁴	5.93
NBP4	3.97x10 ⁷	4.72	4.01x10 ⁶	5.00
Guide Tube 1	5.69x10 ⁷	2.29	2.41x10 ⁶	6.46
Guide Tube 2	6.65x10 ⁷	2.26	3.30x10 ⁶	6.14
Guide Tube 3	6.35x10 ⁷	2.27	4.62x10 ⁶	5.06

Table 7-4: Total gamma flux calculated at the exit of new beam ports and guide tubes in the final PSBR design

Beam Port	Total Gamma Flux (10⁻¹¹ γ/cm²/sec)	Error (%)
NBP1	1.31x10 ⁻¹	10.10
NBP2	8.07x10 ⁻²	11.44
NBP3	1.46x10 ⁰	10.26
NBP4	3.91x10 ⁻¹	10.42
Guide Tube 1	9.12x10 ⁻¹	10.55
Guide Tube 2	9.52x10 ⁻¹	11.28
Guide Tube 3	9.28x10 ⁻¹	10.72

The calculated neutron and gamma flux spectra in the new beam port facilities change based on the beam port configuration including the collimator design, filter material, physical location in the moderator and orientation of the each beam port with the tilt angle with respect to

the core. Since the design feature of each neutron beam port is different, calculated neutron and gamma flux spectra significantly change in the new beam port facilities.

As expected, the highest thermal neutron beam with lowest gamma component was observed in NBP2 because it is directed along the core centerline. Although NBP1 has the same filter and collimator configuration with NBP2, the total thermal neutron flux is 1.6 times lower in NBP1. This is a direct result of the orientation (tilt angle=50°) of NBP1 with respect to NBP2 and its location which is at the periphery of the moderator. On contrary to the thermal neutron beam, the total gamma and fast neutron flux are minimal in this beam port.

In the gamma spectra of NBP1 and NBP2, two additional prompt gamma peaks were observed at 7.631 MeV and 7.646 MeV. These gamma-rays are generated by the thermal neutron capture reactions with ^{56}Fe (91.8% natural abundance) [81]. Since the collimator section has several steel sections in these beam ports, the magnitude of ^{56}Fe gamma-ray peaks are higher than the other observed peaks in the gamma spectra.

As mentioned before, the reactor core has two-fold symmetry. Therefore, NBP4 should produce similar results to NBP1 since they are symmetrically installed to the periphery of the moderator. However, NBP1 has a factor of 1.36 times higher thermal neutron flux, 29 times lower fast neutron flux, and 3 times lower gamma component compared to NBP4. This clearly shows the effectiveness of sapphire crystal to eliminate the fast neutrons and gamma-rays. In addition to the gamma-rays, Bismuth crystal also reduces the thermal neutron content in NBP4. Additionally, 2.2 MeV hydrogen gamma-ray peak is higher than the other peaks observed in the gamma spectrum of NBP4. These gamma rays are generated inside the core where no fuel rods are installed. It should be considered to install the sapphire filter into NBP4.

The neutron flux spectra at the end of NBP3 closely match with the measured spectra in existing BP7 when using the slow-chopper time-of-flight spectrometer [58]. Beside that the total thermal neutron flux is on the same order of magnitude with the measured value in existing BP7.

On the other hand, several gamma peaks at different energies were observed in the estimated gamma spectrum at the end of NBP3. These are mainly generated by the inelastic neutron scattering in ^{23}Na [82] which is in the active content of sandcrete in NBP3 and the ^{58}Ni [83]. The total gamma flux is close to the calculated values in cold neutron source because they are symmetrically installed to the moderator with the same tilt angles of $\pm 20^\circ$.

The MCNP results in the cold neutron beam port are just estimates since the coating material has not been selected yet on the guide tubes. As mentioned before, ^{58}Ni was used as a reference coating material in order to get an estimate about the total cold or total thermal-equivalent neutron flux and total gamma flux. In the neutron spectra, spectral shift from thermal to cold region was clearly observed in the guide tubes. Additionally, the total thermal-equivalent neutron flux values in guide tubes are higher than the ones calculated in NBP1. However, total fast neutron flux is a factor of 17 times higher in Guide Tube 1, 23 times higher in Guide Tube 2, and 33 times higher in Guide Tube 3 compared to NBP1. This is expected because the neutron flux spectrum was calculated at the exit of biological shield and no curvature was defined to the guide tubes. Additionally, the diameter of the cold neutron source was 20 cm at the re-entry hall and the length of the guide tubes was less than 3 meters in the MCNP model. The total fast neutron flux estimated in the guide tubes and NBP4 are on the same order of magnitude with the total fast neutron flux calculated in existing BP4. Additionally, the calculated total gamma flux in guide tubes are on the same order of magnitude with the total gamma flux in NBP3 due to half symmetry of the core.

The lowest utilized total thermal or cold neutron flux in each method and facility was previously given in Table 7-1. MCNP results show that new PSBR design will provide higher thermal or cold beam to the new facilities compared to the tabulated results in this table.

Neutronic Performance of the New Beam Ports Designs Compared to Existing Design

The neutronic performance of the new beam ports were compared to the BP4 in the existing design. The neutron and gamma flux spectra calculated at the end of new beam ports were previously given in Section 5.3. The neutron and gamma flux spectra at the end of BP4 in the existing design was recalculated here again with the depleted fuel compositions in core loading 53H and new cross section data set was employed in the MCNP analysis of the new PSBR design. The same fuel temperatures in all fuel and control rods determined by making a linear interpolation between the measured fuel temperatures in I-16 rod and the calculated local power distribution inside the core was implemented to the MCNP model in this analysis. The neutron flux spectrum calculated at the end of BP4 is shown in Figure 7-21.

The total thermal neutron flux at the end of BP4 was determined by integrating the area under the neutron flux spectrum below 1 eV as

$$\Phi_{ex,thermal} = 3.24 \times 10^7 \pm 6.63 \times 10^4 \frac{n}{cm^2 - sec}. \quad (7-4)$$

The total fast neutron flux (>0.1 MeV) was determined as

$$\Phi_{ex,fast} = 2.79 \times 10^6 \pm 3.98 \times 10^4 \frac{n}{cm^2 - sec}. \quad (7-5)$$

The MCNP estimated total thermal flux at the end of BP4 is within the acceptance range of error ($\pm 10\%$) in measured value (3×10^7 n/cm²/s) by slow-chopper time-of-flight spectrometer [58]. The MCNP predicted thermal neutron flux spectrum was then compared to the measured spectra and to the theoretical Maxwell-Boltzmann energy spectrum model. It is shown in Figure 7-22. The thermal neutron flux spectrum stays within the error bars of the measured spectra.

By using the same analogy, the total gamma flux at the BP4 exit was determined as

$$\Phi_{ex,gamma} = 9.34 \times 10^{-11} \pm 1.19 \times 10^{-13} \frac{n}{cm^2 - sec}. \quad (7-6)$$

The neutronic performance of new beam port designs was then calculated by comparing the total neutron flux and total gamma flux at the end of new beam ports with the results given in Equations (7-4), (7-5), and (7-6) for existing BP4.

NBP3 in the new PSBR design is the same design to BP7 in the existing design. Therefore, the neutronic performance of this beam port should be compared with the existing BP7. As mentioned in the previous chapter, the simulation and measurement results are very close to each other in this beam port. No significant change in neutron flux was observed in NBP3. Similar to NBP3, total fast neutron flux and total gamma flux in PSBR cold neutron source were not considered in the comparison because there will be a curvature in each guide tube to discriminate the fast neutron and gamma-rays in the actual design. The neutronic performance of new beam ports is given based on the increase or decrease rate in the thermal and fast neutron flux and gamma flux compared to existing BP4. The results of this analysis are given in Table 7-5. Additionally, the thermal-to-fast neutron ratio calculated in the new beam ports are given in Table 7-6.

The comparison of the MCNP calculated results between the new beam ports and the existing BP4 shows that the best performance was obtained in NBP1 which had the highest thermal-to-fast neutron flux ratio with a very low gamma component. On contrary, the worst neutronic performance in the new beam ports was observed in NBP4 even if it is symmetrically installed to the edge of moderator with NBP1. This is expected because NBP4 has the same design characteristics with the BP4 in the existing PSBR design. But, the thermal-to-fast neutron flux ratio in NBP4 is ~10 times higher and gamma contamination is 238 times lower in NBP4 compared to the existing BP4 due to the new core-moderator assembly design.

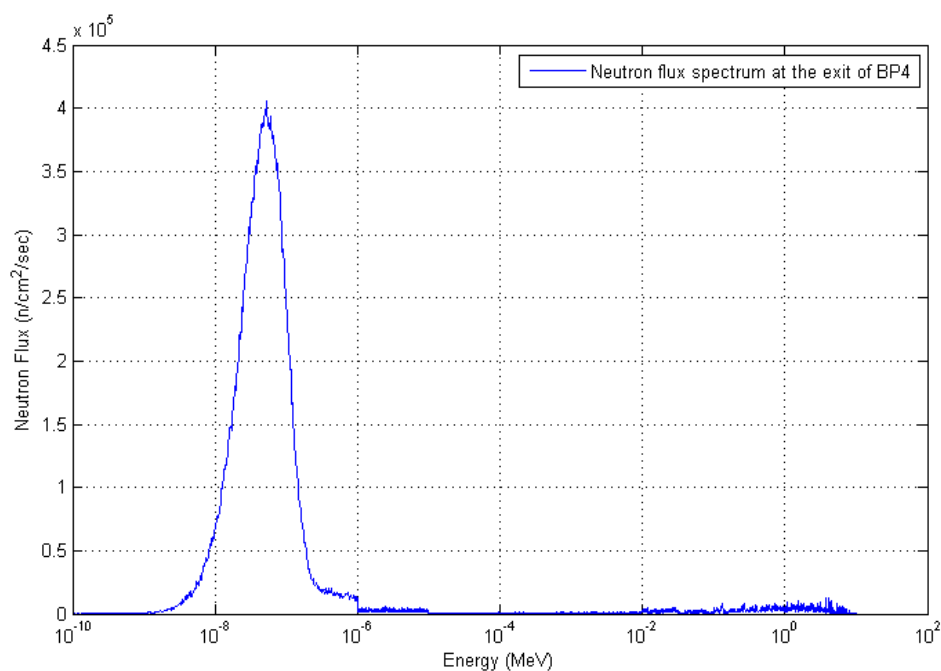


Figure 7-21: The neutron flux spectrum at the end of existing BP4 calculated by MCNP model with depleted fuel and control elements of core loading 53H (May 2009)

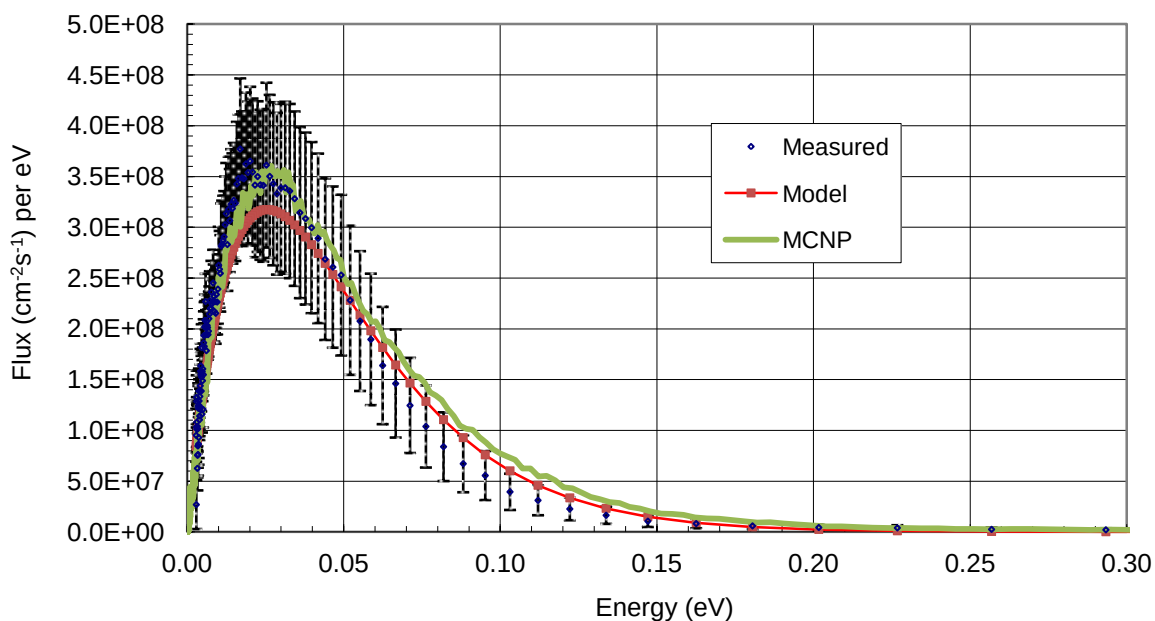


Figure 7-22: Comparison of the MCNP predicted thermal neutron flux spectrum at the end of existing BP4 with the measured neutron flux spectrum with slow neutron chopper TOF spectrometer and the theoretical Maxwell-Boltzmann energy spectrum

Table 7-5: Neutronic performance comparison between new beam ports in the final PSBR design and BP4 in the existing design

Beam Port	$\frac{\Phi_{thermal,new}}{\Phi_{thermal,existing}}$	$\frac{\Phi_{fast,new}}{\Phi_{fast,existing}}$	$\frac{\Phi_{gamma}}{\Phi_{gamma,existing}}$
NBP1	1.68	0.05	0.01
NBP2	2.68	0.32	0.0086
NBP4	1.23	1.44	0.0042
Guide Tube 1	1.75	0.87	NA
Guide Tube 2	2.05	1.18	NA
Guide Tube 3	1.96	1.65	NA

Table 7-6: Neutronic performance of new beam ports based on the thermal-to-fast neutron ratio

Beam Port	$\frac{\Phi_{thermal}}{\Phi_{fast}}$
NBP1	383.94
NBP2	94.24
NBP4	9.88
Guide Tube 1	23.54
Guide Tube 2	20.13
Guide Tube 3	13.74

NBP2 has the highest total thermal neutron flux in all the new beam ports as shown in Table 7-5. On the other hand, the thermal-to-fast neutron ratio in NBP2 is lower than the ratio calculated in NBP1 since the number of fast neutrons in NBP2 is ~4 times higher compared to NBP1. However, total fast neutron flux is ~3 times lower in NBP2 compared to existing BP4. Therefore, the overall performance of this beam port is significantly better than the one in BP4.

For the PSBR cold neutron beam port, only the total thermal-equivalent neutron flux will give a clue about the neutronic performance of the guide tubes. The fast neutrons and gamma-rays will be diminished in the cold neutron beam facilities after the guide tubes will have

curvature. As seen in Table 7-4, ~2-times higher thermal-equivalent neutron flux was observed in the guide tubes compared to the existing BP4.

Although different results were calculated in the neutronic performance of new beam ports, significant neutronic performance improvements were observed in the new proposed facility compared to the existing PSBR design.

Chapter 8

SUMMARY, CONCLUSION, AND FUTURE WORK

Existing core-moderator assembly and beam port design limits the experimental capability of the Penn State Breazeale Reactor (PSBR) of the Radiation Science and Engineering Center. The problems in the existing design were identified as

- the number of beam ports that can be simultaneously utilized in the facility,
- contamination of the neutron beam by hydrogen gamma-rays in the current utilized beam port facilities.

In this study, a new core-moderator assembly and five new beam ports were designed to overcome these inherent design issues. A crescent shaped moderator shape was selected in the new assembly. Then, a proper coupling was achieved between the core and crescent moderator tank by minimizing the pool water around the reactor core and in the new core-moderator interface. Furthermore, the dimensions of the top and bottom grid plates in the current design were changed to promote the coupling. The reactor tower that holds the reactor core together was redesigned by installing four new support bars which were welded to the grid plates to support the core. New core-moderator assembly can simultaneously utilize up to five new beam ports. The locations and relative orientations of new beam ports with respect to the core-moderator assembly were determined by evaluating the available floor space in the beam hall and location of the core in the pool.

Neutronic analysis of the new PSBR core-moderator assembly and new beam ports was achieved by making MCNP simulations in three subcategories. First the excess reactivity of the new core design was calculated. Fifteen-cent gain in the excess reactivity was determined in the new core-moderator assembly compared to the existing design. Beside that the contribution of the

new moderator tank to the excess reactivity of the system was calculated as \$1.13 which is significantly higher than the excess reactivity contribution of existing D₂O tank, \$0.68. These analyses were performed for reference core loading 53H of the PSBR. The depleted fuel compositions in this core loading were calculated by TRIGSIMS code [17]. The temperatures of the fuel sections in all the fuel rods were set to 800 K and a burnup and temperature dependent cross section data set generated by Tippayakul [56] was employed during these calculations.

In the second section, the optimum design was evaluated by applying a state-space search approach. First, the design parameters in the optimization study were determined by considering the design requirements and design constraints. Design parameters were categorized as constant, search and control parameters. The constant parameters were identified by considering the design requirements and constraints. The height of the moderator was set to 48.2 cm in order to increase the cross-flow into the core from 5.2-cm left between moderator and the top and bottom of the moderator tank. The locations of the new beam ports with respect to the pool floor were defined as the center of fuel rod about its active length. Search parameters or the optimum design parameters were identified as the radius of the crescent shaped moderator tank and the distances between the core and the new beam ports. The control parameters were identified as thermal-to-fast neutron ratio, total thermal neutron flux and total gamma dose in the new beam port facilities. The purpose of the optimization study is to calculate optimum design parameters or the search parameters which maximize the thermal-to-fast neutron ratio and total neutron flux with minimum gamma dose in the beam port facilities. MURE code was implemented in these calculations to generate MCNP inputs. All the beam ports were modeled as single empty tubes with 15.2-cm outer diameter. The detailed information about this analysis is given in Chapter 5. The optimum size of the crescent moderator tank in the new design was calculated as 76.2 cm (30 inches) in radius and optimum distance values between new beam ports and the core face was found as 15 cm for NBP2, 18 cm for NBP3 and 12 cm for NBP4, respectively. Since the core has

two-fold symmetry, the optimum distance values for NBP1 and cold neutron BP were assumed to be same with calculated values for NBP4 and NBP3, respectively.

MCNP simulations in the optimization study were achieved by two-steps calculations with “full core” and “beam port” models. In the full core model, a criticality calculation using KCODE card was performed to fabricate a surface source at a certain depth of each beam port. The angular and spatial distributions of incident neutron and gamma beam were calculated on the source surface. In the beam port model, neutrons and gammas were sampled on this surface by defining the angular and spatial distributions in general source card (SDEF). Then, the neutron and gamma spectra at the end of each beam port were determined. This calculation approach converted the criticality problem into a fixed source problem and significantly decreased the computational time. Moreover, two variance reduction techniques were implemented to increase the statistical accuracy and to decrease the relative error in the tallies. These are conical source direction biasing technique and the use of point detector estimators. The conical source direction biasing technique was utilized to define the directions of each particle on the surface source. The directions of source particles were calculated by using the angular and spatial distribution of the incident beam on this surface. They are restricted to a set of cones in this technique. The probability of neutron emission in any direction or angular bin ($-1 < \mu = \cos\theta < 1$) were defined using SP and SB cards in the MCNP inputs. On the other hand, point detectors were used to transport more source particles to the tally regions located at the end of the beam ports. It gives an estimate of the particle flux in space.

The accuracy of this calculation approach was verified by comparing the MCNP results with the measurements obtained with slow neutron-chopper time-of-flight (TOF) spectrometer [58] measurements performed at the end of BP4 in the existing PSBR design model.

In the final part of the optimization study, the neutronic performance of the new-core moderator assembly with the calculated optimum design parameters was determined. This was

achieved by comparing the MCNP calculated total neutron flux and total gamma flux in new beam port facilities with the measurements. The same beam port model, BP4 in the existing facility, was utilized to the entire new beam port models during the calculations as a reference beam port and for comparison. The calculation methodology was described in Chapter 3 and 5. The results showed a significant neutronic performance improvement in the new facility. Compared to the current beam port facility, the total thermal neutron flux was almost 2 times higher and total gamma dose was reduced by 77% in the new beam port facilities. This clearly showed the neutronic performance improvement in the new PSBR design due to utilization of the new core-moderator assembly design. Additionally, a considerable increase in the total fast neutron flux was observed in the new beam port models compared to the existing BP4 as a result of the MCNP calculations. Since all new beam ports were directed to core center, it was expected to see an increase in the fast neutron flux as well as the thermal neutron flux. Furthermore, the optimum design parameters were selected by considering the increase in thermal neutron flux and decrease in gamma flux. The fast neutrons in the new beam port were expected to be eliminated by using an additional effective filter materials e.g. hemlite-grade sapphire filter. The increase in the new beam ports modeled using existing BP4 design in the optimization study clearly shows the inefficiency of the Bi crystal to filter most of the fast neutrons in BP4.

The thermal-hydraulics performance of the new core-moderator assembly was evaluated by making a computation fluid dynamics (CFD) simulation in the ANSYS Fluent (v14.0) environment. The primary purpose of the study was to calculate the temperature and velocity profiles in the new core-moderator assembly, in order to verify the safe operation of the reactor. The safety limits set the maximum fuel temperature to (1) 1150 °C in steady-state and (2) 950 °C during LOCA in the PSBR. The clad outer temperature does not exceed 500 °C under these conditions.

The CFD simulation was performed in reference core with loading 53H. Each fuel rod in this core has different heat generation rate as a function of burnup. The heat generation rates in the fuel rods were estimated by TRIGSIMS burnup code [56]. Conjugate heat transfer (CHT) modeling approach which solves the fluid and solid equations simultaneously by coupling the heat transfer rate from fuel region to coolant was not considered in the simulations because of the extensive computational mesh and computational time requirements. Instead, the amount of heat generated by fuel was assumed to be transferred into the coolant and surface heat flux was applied to claddings' outer surface by considering the burnup level of the individual fuel rod in reference core loading 53H at 1MW full power. The surface heat flux was determined by using the local power of each individual fuel rod estimated by TRIGSIMS code.

Approximately, 22 million structured and unstructured meshes were generated in the core and the pool in order to obtain accuracy and convergence in the CFD solutions. Natural convection was modeled with Boussinesq approximation. Buoyancy driven flow in the flow channels was assumed to be turbulent except at the lower parts of fuel rods where there was no heat generation. The turbulent model, $\kappa - \omega$ with SST formulation [59], was implemented to model the laminar to turbulence transition. Subcooled nucleate boiling was modeled with Eulerian-Eulerian multiphase flow formulation and the RPI wall boiling model [61]. The RPI model defines the wall heat flux partitioning from the clad outer surface to subcooled flow. The detailed information about the modeling of the multiphase flow in new core-moderator assembly was discussed in Chapter 6.

The thermal-hydraulics performance of the new core-moderator assembly design was evaluated by comparing the Fluent CFD code predicted bulk fluid temperatures with the measured bulk fluid temperatures in the hot channel of loading 53H (2009) and four flow channels in the PSBR core by Haag [23] (1971). Additionally, Cobra-TF (CTF) simulation results obtained in the existing PSBR were also utilized in the comparison. Bulk fluid temperatures in

loading-53H were measured and calculated at the middle of the hot channel: (1) in the center of the core and (2) 12.7-cm above the core center.

The CFD results were given first for the axial and radial flow paths at the center of the core. The first observation in these results is the secondary radial flow entering into the core due to decrease in the cross-flow. Additionally, the radial flow improves the local heat transfer conditions by providing radial-mixing in the core. For the axial flow redistribution along the flow channels, reasonable flow symmetry has been observed. Focusing on the bottom grid plate, orifice effect and flow development along the channels were observed. Flow develops from laminar velocity profile to turbulent as a result of the heat transfer to the coolant in the flow channel. Velocity magnitudes in the hot channel are between 0.12 m/s and 0.14 m/s. The bulk fluid temperature is calculated as 32 °C at the beginning of active fuel length and it increases up to 88 °C at the channel exit. In contours of void fraction on the heated fuel rods in the hot channel, nucleation forms a bubble layer on the fuel rods. It spreads a few layers in the thermal boundary but bubbles then collapse in the subcooled fluid. This does not change the bulk fluid properties.

The CFD calculated temperature results are significantly different than the measurements in core loading 53H. A temperature difference of 15 °C was observed between the simulation and experiments such that CFD results are higher. Two possible problems were outlined in Chapter 6 to explain this temperature difference between the results. The temperature measurements were conducted by using a thermocouple in an aluminum tube with a considerable size with the flow area of the channel. The aluminum tube causes flow mixing in the flow channel. Therefore, the velocity of the coolant at that channel should increase by improving the heat transfer conditions. Another possibility is the decrease in the cross-flow by the new crescent moderator tank. This decrease will give a rise in the bulk fluid temperature. Although the temperature rise is not negligible, the bulk fluid temperature is estimated below the saturation temperature (111.4 °C) of

the fluid at operating pressure of 1.5 atm. This verifies that subcooled nucleate boiling did not shift to nucleate boiling in the new PSBR core. Compared to Cobra-TF simulation of the new PSBR design, ~ 11 °C temperature rise was observed in the Fluent CFD calculated bulk fluid temperature profile in the hot channel of the new PSBR core.

The comparison between the CFD simulation results obtained at the new PSBR core and bulk fluid temperature measurements conducted in the PSBR core (1971) supported the expectations about the effect of cross-flow on the cooling of the core. This comparison was achieved in four flow channels named as L, M, N and P starting from the core center to the edge. During the measurements, a decrease in the bulk fluid temperature profile was reported above 2/3 of active length of the fuel rods in the flow channels. The temperature decrease was higher in the flow channels close to the edge where the cross-flow rate was higher. On the other hand, the bulk fluid temperature is continuously increasing along the flow channels of the new PSBR core. This clearly verifies that the cross-flow is significantly reduced by the new moderator tank and the axial-flow will be effective on the cooling of the core in the new design.

The maximum fuel temperature in the hot channel of the new core was determined by using the maximum clad outer temperature estimated as 174.85 °C in the hot channel by the CFD calculations. For the fuel and clad thermal conductivity and gap heat transfer coefficient, empirical correlations suggested by General Atomics (GA) [74] for TRIGA fuel were employed in these calculations. The maximum fuel temperature in the hot channel was estimated as 482 °C, which is different than the measured fuel temperature (540 °C) in the existing reactor. The GA correlations for the gap convective heat transfer coefficient and the fuel thermal conductivity is the result of this difference since the burnup dependence of these properties is underestimated in the correlations. A higher fuel temperature which will be close to measured result is expected in the calculated results. Therefore, both the maximum fuel temperature and the maximum clad

outer temperature estimated by the ANSYS Fluent CFD analysis validate the thermal-hydraulics safety of the new PSBR design.

After an optimum core-moderator assembly design was evaluated and the thermal-hydraulics safety of the optimum design was verified, a final analysis was performed to calculate the neutron and gamma flux spectra in five new beam port designs. In the existing facility, almost all the experiments have been conducted in BP4. After the design change, seven different experimental facilities can be simultaneously utilized in the new facility. It was considered to add five new experimental methods to the existing facilities which are Neutron Imaging and Neutron Transmission facilities. New methods were selected as Triple-Axis Spectrometer (TAS), Prompt Gamma Activation Analysis (PGAA), conventional Neutron Depth Profiling (NDP) and Time-of-Flight (TOF)-NDP, and Neutron Powder Diffraction (NPD). One of the thermal beam ports will be reserved for explanatory experiments. Each one of these neutron beam techniques and experimental facilities mentioned above will be utilized in a specific beam port. Therefore, the beam port designs in the new facility were selected based on the requirements of experimental methods or facilities [21]. Design features and characteristics of each beam port and guide tube were given in Chapter 7.

The neutronic analysis of the new beam ports were achieved by MCNP simulations performed in two-stages with conical source direction biasing and point detector estimator variance reduction techniques. Core loading 53H at 1MW power was taken as a reference core model. The depleted fuel compositions calculated by the Burned Coupled MCNP simulation tool developed for the PSBR [79] were employed in these calculations. Additionally, a cross section data set generated for 85 isotopes with 10 °C temperature increments between 293 K and 900 K was used. The fuel temperatures in each fuel rod was estimated by making a linear interpolation between the local power generated at each axial section of each fuel and control rod and the measured fuel temperature at instrument rod 16 (I-16).

MCNP simulations in New BP1 (NBP1) and New BP2 (NBP2) were achieved with a different calculation methodology since there is no available cross section data set in the evaluated tables generated for the hemlite-grade sapphire filter. The neutron transmission rate in the 7.62-cm thick hemlite-grade sapphire filter was estimated by employing an empirical correlation named Cassels function [44], which defines the neutron attenuation cross section as a function of energy. The incoming neutron beam on the sapphire filter in NBP1 and NBP2 was calculated in “full core” model. Then, transmitted neutron beam in the sapphire filter was calculated through the Cassels function. Thermal neutron transmission rate was estimated as ~74% and fast neutron attenuation rate was ~78% as a result of these calculations. The estimated thermal neutron transmission rate closely matches with the measured value by Dr. Ünlü. However, the fast neutron attenuation rate is below the measured value (90%). This was expected because of the reported discrepancies for the Cassels function in the fast neutron energy region by Mildner [44]. The neutron and gamma spectra at the end of NBP1 and NBP2 were then calculated by MCNP simulations in the “beam port” model with conical source direction biasing and point detector estimator. It was assumed that transmitted neutron beam in the sapphire filter will have the same source biasing parameters with the incoming neutron beam to the entrance surface of the filter.

For the PSBR cold neutron source, the neutron guide tubes were modeled with ^{58}Ni coating material and up to exit of biological shield exit. The neutronic performance of the cold neutron source was evaluated under these conditions.

The calculated neutron and gamma flux spectra at the end of new beam ports and neutron guide tubes in the final design were given and discussed in Chapter 7. Important outcomes about these analyses can be summarized as follows. The neutronic performance of each new beam port is directly related to its geometrical configuration in the core-moderator assembly and the filter material and collimator system. The highest thermal neutron flux was observed in NBP2.

Although total gamma flux will be smaller in all new beam ports compared to the gamma flux in the existing BP4, NBP1 and NBP2 have the smallest gamma component due to effective filtering by the 7.62-cm thick hemlite-grade sapphire crystal. On the other hand, NBP4 had the highest fast neutron beam component because of its filter design. This was consistent with the observations discussed in Chapter 5. The total gamma flux in NBP4 is 0.42% of the total gamma flux in existing BP4.

Conclusion

It is demonstrated computer simulations and verified by experimental measurements that the new PSBR design proposed in this study provides several improvements by enhancing the experimental capability of the existing facility. The new design is aimed to solve the inherent design issues while preserving the advance characteristics of the existing reactor. The conclusions of this study can be summarized as follows.

This study identified all the limitations of existing reactor and proposed proper solutions to each one of them. The proposed reactor design was built based on this demand. The major limitations of the reactor were identified as the core-moderator assembly and geometrical configuration of the neutron beam ports with respect to core-moderator assembly. The prompt gamma component of the neutron beam in BP4 is directly to them. These limitations were previously investigated in several studies [18][21][22][84] and minor improvements were suggested and demonstrated experimentally and computer simulations to improve the neutronic performance of the existing BP4. However, core design has never been addressed in any of these studies. Although the neutron beam ports are coupled with the reactor core through the moderator tank, the coupling of the core and moderator tank is the key issue to increase the number of available neutron beam ports in the facility and to eliminate the hydrogen gamma contamination

of the neutron beams in the beam ports. The proper coupling can only be achieved by modifying the core design through the top and bottom grid plates and the shape of the moderator tank. In this study, a crescent-shaped moderator tank is proposed and coupled with the new grid plates which are equal and smaller in size than the existing grid plates. The tower design is also necessarily being changed to support the new reactor core by adding four support bars and two support plates on top of the new core. These modifications allow the simultaneous utilization of five new neutron beam ports in the experimental facilities with suppressed gamma dose. Moreover, the core will still be movable after the design changes.

Five new neutron beam port designs are introduced in this study and a cold neutron beam port with a similar design to University of Texas Cold Neutron Source (UTCNS) is added to the facility. The proposed cold neutron beam port will provide cold neutrons from three curved guide tubes. In the new facility, each new beam port and also each guide tube is utilized to a single experimental technique and facility. Four new experimental techniques in addition to existing ones are proposed to be utilized in the new beam hall for this purpose. Therefore, new neutron beam ports are designed in order to accommodate the requirements of experimental technique. Additionally, the number of neutron beam ports will be seven as in the existing facility.

In this study, the modeling of the proposed PSBR design was achieved with highly detailed neutronic simulations by employing Monte Carlo Utility for Reactor Evolutions (MURE), MCNP, and two burnup codes developed for the PSBR: TRIGSIMS [17] and Burned Coupled Simulation Tool for the PSBR [79]. The optimum design parameters and the neutronic performance of the new beam ports were successfully determined by employing these codes. The burnup codes calculate the depleted fuel compositions in the PSBR fuel and control rods. They are not capable of making beam port calculations which are beyond the scope of these codes. Thereafter, burnup codes were employed to characterize the fuel and control rods for the beam port calculations. The MURE code was employed to automatically generate MCNP inputs for the

optimization study. The neutronic modeling of the new PSBR design couples the burnup calculations achieved with aforementioned two codes with the beam port calculations performed with MURE and MCNP codes.

The neutronic simulations of the beam ports in this study suffered from the deep penetration problem. This problem was successfully overcome by making simulations in two-steps with conical source direction biasing technique and point detector estimators. This simulation approach is significantly shortened the computational time and the accuracy in the tally results. Furthermore, the accuracy of the simulation approach was verified by the measured neutron spectra obtained with slow neutron chopper time-of-flight spectrometer in the existing BP4 [58]. Additionally, the neutron transmission in single-crystal hemlite-grade sapphire filter and the new neutron beam ports were successfully modeled with measured transmission cross sections and an empirical correlation, named Cassels function. The simulation results match to the measured values with reasonable accuracy.

In the case of simulations, the optimization of the new core-moderator assembly and beam port designs are based on the maximization of the thermal-to-fast neutron ratio and total gamma dose in the beam port facilities. The maximization of thermal-to-fast neutron ratio in the beam ports is achieved by increasing the rate of thermalization in the moderator tank and estimation of the optimum distance between each new beam port and the core faces. The gamma contamination problem is diminished by directing the beam ports to the core center.

The design features of the new PSBR not only solve the limitations of the existing design but also provide significant neutronic performance increase in the beam port facilities. As a result of the MCNP simulations, a minimum of 1.23 times (NBP4) and a maximum of 2.68 (NBP2) higher thermal neutron flux are expected in the new beam ports compared to the existing BP4. Besides, the gamma components of the neutron beams in the new beam ports are reduced by a factor of 100. These results clearly indicate the neutronic performance improvements that can be

attained in the new beam port facilities with the utilization of the new core-moderator assembly and new beam port designs with the estimated optimum design parameters.

In this study, the thermal-hydraulics performance was investigated to assure the safety limitations in the new core-moderator assembly. The detailed geometrical modeling was obtained by CFD modeling of the new PSBR design. Furthermore, the reduction in the cross flow and the subcooled boiling under natural convection conditions were analyzed by investigating flow paths and bulk fluid velocity and temperature profiles in the new core-moderator assembly design in a very detailed manner. Utilization of CFD simulations revealed crucial information on thermal-hydraulics performance of the new core-moderator assembly as follows:

- The reduction in the cross flow increased the bulk fluid temperature
- The decrease in the bulk fluid temperature profile above core center in the existing design was not observed in the new design. This clearly indicated that the loss of cross-flow had significant effect on the cooling of the core.
- Bubble formation was observed on the fuel cladding outer wall. However, bubbles did not alter the bulk fluid velocity profile in the flow channels. Bubbles which detached from the wall were collapsed in the thermal boundary layer. Bubbles only enhanced the heat transfer from cladding to the bulk fluid.
- The bulk fluid temperature was below the saturation temperature of the water at the PSBR operating conditions.
- The nuclear safety criterions were met by the new optimum design of the PSBR.

This study proves the accomplishment of our “unique target” which is to overcome the limitations of the existing PSBR and to expand the experimental capability of the beam port facilities. Additionally, it is shown that the new PSBR design will operate under thermal-hydraulics safety limits.

The accomplishment of the design for the new core-moderator assembly combined with the new beam ports and neutron beam techniques will bring significant academic and economic benefits to the Penn State faculty and students in many disciplines. Neutron beam facilities will provide excellent research opportunities for development of instruments and experiments to the researchers at national and regional universities, industry, and national laboratories. Hence, Radiation Science and Engineering Center will be a complete, modern, and advanced research facility.

Future Work

This study provides several unique features from thermal-hydraulics and neutronic point of view of the new PSBR design and can be extended and improved with additional studies. The suggested future works are given as follows.

CFD simulations of the new PSBR design showed the bubble nucleation forming on the heated fuel rods. However, the simulation results indicated that the temperature and velocity profile of the bulk fluid was not altered by the presence of bubbles. Bubbles only enhanced the heat transfer from the wall. This is mainly due to improvement of convective heat transfer coefficient for the water due to subcooled nucleate boiling. The size of the computational meshes near the heated walls is a very effective factor for the formation and detachment of bubbles. Additionally, the mesh size close to the heat walls should also be independent from the wall heat flux partitioning to the liquid and vapor phases. As a future work, another study on relationship between near wall mesh size and the bubble nucleation sizes and between near wall mesh size and wall heat flux partitioning model is suggested. This can be very effective to understand the bubble nucleation dynamics in the PSBR.

The Fluent CFD simulation results for the bulk fluid temperature in the hot channel were compared with the measured values in core loading 53H. The measurements were conducted with a thermocouple installed into an aluminum tube, which was inserted into the hot channel during the experiment. The size of the tube is considerable to the flow area. Thus, the aluminum tube decreases the flow area in the flow channel and changes the actual flow behavior. Thus, the measured bulk fluid temperatures do not represent the actual flow conditions. It is suggested to figure out and employ a better technique for the measurement of the bulk fluid temperature. However, it is not possible to repeat the temperature measurements under same operating conditions since the core loading 53H was recently changed to core loading 54, which has a significantly different loading pattern than the loading pattern of 53H. Therefore, there are two options: the ANSYS CFD analysis should be performed (1) for the existing reactor design with core loading 53H or (2) for the new reactor design with the core loading 54. By considering the success of the CFD calculation methodology demonstrated in this study, as a future work, it is suggested to use first option. This will also be useful to understand the operating conditions of the existing reactor.

The simulation approach in the CFD analysis was to apply the heat generated by each fuel rod to the water coolant by ignoring the conduction heat transfer inside the fuel rod. Although this approach was correct under steady state operating conditions of the reactor, the CFD analysis can be further advanced by modeling the solid regions in the calculation domain via the conjugate heat transfer (CHT) approach. A coupled solution in the fluid and solid domains can be achieved and then temperature profile in the fuel rods can be calculated by using CHT approach. The only problem in the CFD modeling of the fuel rods is the lack of reliable data for the thermal properties of the U-ZrH_x fuel and gap. The effect of the gap size on fuel temperature was previously investigated by Karriem [28] in sensitivity studies. The gap size was estimated as 0.0025 cm in the fuel rods and this size can be used in the CFD modeling with CHT approach. As

an initial guess, the General Atomics suggested correlations [74] can be used to define the thermal properties of the fuel and the gap. As a future work, these properties can be estimated in an iterative calculation by using predictor-corrector numerical methods. In the predictor step, the fuel thermal properties will be estimated using the suggested values by General Atomics and then they will be updated based on the difference between the CFD predicted and measured fuel temperatures in the corrector step. If the bulk fluid temperature can be precisely measured in the existing reactor, it can even be employed to decrease the numerical errors. The accurate calculation of the thermal properties of fuel conductivity and gap heat transfer coefficient in TRIGA fuel rods will improve the CFD results. The calculated temperature distribution inside the fuel rods can then be provided for the PSBR's burnup codes (TRIGSIMS [17] and Burned Coupled Simulation Tool for the PSBR [77]) as a temperature feedback. In addition to this, void-reactivity feedback can be calculated by the CFD predicted local void distribution in the near wall region of flow channels [85]. Both the void-reactivity feedback and temperature feedback will improve the burnup calculations.

Another future work could be addition of N-16 pump with nozzles to the CFD model. Although, an N-16 pump is not significantly affecting the velocity and temperature profile of the bulk fluid inside the core, it can improve the simulation results in the reactor pool side.

The design of the new PSBR core is based on the loading 53H. However, the core loading was recently changed to a new loading pattern, named loading-54. The fuel and control rod positions in these loading patterns are significantly different than each other. Therefore, the utilization of the loading-54 pattern into the new core design needs to be evaluated.

The performance of the 7.62-cm thick hemlite-grade sapphire filter that will be employed in new beam ports were estimated by using Cassels function based on the neutron transmission measurements at NIST [44]. Although reasonable accurate results were attained with this

approach, the cross section generation for the sapphire filter should be considered to decrease the computational time and complexity.

The modeling of the cold neutron beam port and the guide tubes was made until the biological shield exit and ^{58}Ni coating material. It is necessary to analyze this beam port after all the design parameters, which are the coating material, length of the guide tubes, and the curvature applied on the guide tubes, were selected. It is also required to analyze the neutronic performance of NBP4 with an additional filter material, such as sapphire crystal, in order to eliminate the fast neutrons in this beam port.

The neutronic analysis of the entire, existing and new, PSBR design; has revealed the optimum design parameters and demonstrated the performance improvement that can be attained in the new proposed beam port facilities. The MCNP modeling approach demonstrated in this study was verified with the experimental results in the existing BP4 and it can be further utilized in all the beam port simulations of the new improved PSBR. The implementation of this approach to the TRIGSIMS code [17] or Burned Coupled MCNP Simulation Tool [77] would enhance the modeling capability of these codes. By coupling one of the above mentioned codes with ANSYS's Fluent CFD model; an advanced modern computation tool that can satisfy all the modeling needs for the new PSBR can be attained.

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Appendix A

Conservation Equations for Single Phase Flow with Heat Transfer

ANSYS Fluent and CFX codes solve the momentum and mass conservation equations to model fluid flows with single phase. An additional equation for energy conservation is solved for flows involving heat transfer. Mathematical formulations of these equations were described in the following subsections.

Continuity Equation

In general, the conservation equation for mass or continuity equation can be written as follows.

$$\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho \vec{v}) = 0 \quad (\text{A-1})$$

where ρ is the density of fluid and $\vec{v}$ is the phase velocity vector. The first term on the left side of the equation represents the density change in time and the second term is the density change in space.

Momentum Equation

Conservation of momentum can be written as follows.

$$\frac{\partial}{\partial t} (\rho \vec{v}) + \nabla \cdot (\rho \vec{v} \vec{v}) = \nabla p + \nabla \cdot \bar{\bar{\tau}} + \rho \bar{g} + \bar{F} \quad (\text{A-2})$$

where p is the static pressure, $\bar{\bar{\tau}}$ is the stress tensor, and $\rho \bar{g}$ and $\bar{F}$ are the gravitational body force and external body forces, respectively. The stress tensor $\bar{\bar{\tau}}$ is given by

$$\bar{\bar{\tau}} = \mu \left[(\nabla \vec{v} + \nabla \vec{v}^T) - \frac{2}{3} \nabla \cdot \vec{v} \mathbf{I} \right]. \quad (\text{A-3})$$

In Equation (A-3), μ is the molecular viscosity, $\mathbf{I}$ is the unit tensor, and the second term on the right hand side of the equation is the effect of volume dilation. The terms on the left hand side of the Equation (A-2) represent the momentum change in time and space, respectively.

Energy Equation

Fluent solves the energy equation to describe the heat transfer to the fluid and/or to solid.

The energy equation is given as

$$\frac{\partial}{\partial t}(\rho E) + \nabla \cdot (\vec{v}(\rho E + p)) = \nabla \cdot \left(k_{\text{eff}} \nabla T - \sum_j h_j \bar{J}_j + (\bar{\bar{\tau}}_{\text{eff}} \cdot \vec{v}) \right) + S_h, \quad (\text{A-4})$$

where k_{eff} is the effective conductivity and $\bar{J}_j$ is the diffusion flux of species j . The first three terms on the right-hand side of this equation represent energy transfer due to conduction, species diffusion, and viscous dissipation, respectively. In Equation (A-4),

$$E = h - \frac{p}{\rho} + \frac{v^2}{2} \quad (\text{A-5})$$

in which

$$h = \sum_j Y_j h_j. \quad (\text{A-6})$$

In Equation (A-6), Y_j and h_j is the mass fraction and enthalpy of species j , respectively.

Natural Convection or Buoyancy Driven Flows

The ratio of Grashof and Reynolds numbers and the Rayleigh number are used to describe the buoyancy driven flows. The ratio of Grashof and Reynolds number is employed to identify the importance of buoyancy driven forces in a mixed convection flow.

$$\frac{Gr}{Re^2} = \frac{\vec{g}\beta\Delta TL}{\nu^2} \quad (A-7)$$

If this ratio is smaller than unity, buoyancy driven forces are ignored. In pure natural convection, the strength of the buoyancy-induced flows is measured by Rayleigh number:

$$Ra = \frac{\vec{g}\beta\Delta TL^3\rho}{\mu\alpha} \quad (A-8)$$

where

$$\beta = -\frac{1}{\rho}\left(\frac{\partial\rho}{\partial x}\right)_p \quad (A-9)$$

and

$$\alpha = \frac{k}{\rho c_p} \quad (A-10)$$

If $Ra < 10^8$, there is a buoyancy-induced laminar flow. Transition to turbulence occurs over the range $10^8 < Ra < 10^{10}$.

Appendix B

Conservation Equations for Multiphase Flow with Heat Transfer

The mathematical formulations of conservation equations to describe the multiphase flow with heat transfer are given in this section.

Continuity Equation

Continuity equation for phase k is written as follows:

$$\frac{\partial \rho_k \alpha_k}{\partial t} + \nabla \cdot (\rho_k \alpha_k \vec{v}_k) = \sum_{\substack{l=1 \\ l \neq k}} \Gamma_{kl} \quad (\text{B-1})$$

In this equation, the subscripts k and l denotes the phases, α_k is the volume fraction of each phase and Γ_{kl} is the mass transfer between the phases.

Momentum Equation

Conservation of momentum for phase k in two-fluid multiphase flow model can be written as follows.

$$\begin{aligned} \frac{\partial}{\partial t} (\rho_k \alpha_k \vec{v}_k) + \nabla \cdot (\rho_k \alpha_k \vec{v}_k \vec{v}_k) = & -\alpha_k \nabla p_k + \nabla \cdot [\alpha_k (\tau_k + \tau_k^T)] \\ & + \rho_k \alpha_k \vec{g} + M_{ki} - \tau_{ki} \cdot \nabla \alpha_k + (p_{ki} - p_k) \nabla \alpha_k, \end{aligned} \quad (\text{B-2})$$

where interface momentum exchange, velocity, shear and pressure are denoted by M_{ki} , v_{ki} and p_{ki} , respectively. τ_k is the viscous shear stress on phase k [80] and τ_k^T is the bubble-induced

turbulence terms [63]. Momentum exchange between the phases is defined by using the interfacial forces as follows.

$$M_{ki} = \vec{F}_\Gamma + \vec{F}_D + \vec{F}_L + \vec{F}_{WL} + \vec{F}_{TD} + \vec{F}_{VM}. \quad (B-3)$$

In this equation, $\vec{F}_\Gamma$ is the secondary momentum transfer force, $\vec{F}_D$ is the drag force, $\vec{F}_L$ is the lift force, $\vec{F}_{WL}$ is the wall lubrication force, $\vec{F}_{TD}$ is the turbulence dispersion force and $\vec{F}_{VM}$ is the virtual mass force.

In ANSYS Fluent, interphase momentum transfer in bubbly flow were modeled with four interfacial forces: drag force, lift force, turbulent dispersion force and virtual mass force [87] [88].

The velocity of each phase in the flow is different. Liquid is slower than the vapor due to density difference. Drag force tends to slow down the vapor and speed up the liquid phase. Almost all definitions of drag force included drag coefficient (C_D) which is based on the Reynolds number. There are five drag force models available under the RPI wall boiling model. They are boiling-Ishii, universal drag, Schiller-Naumann [89], Symmetric and Morsi-Alexander models . Detailed information about these model are given in ANSYS Fluent theory guide [73]. Schiller-Naumann drag force model was employed in this study. This model is applicable to all fluid-fluid flows. In this model, the drag force between the phases is defined as

$$\vec{F}_D = \frac{C_D Re}{24} \quad (B-4)$$

where

$$C_D = \begin{cases} 24 (1 + 0.15Re^{0.687}) & Re \leq 1000 \\ 0.44 & Re > 1000 \end{cases} \quad (B-5)$$

The lift coefficient for the lift force on the liquid phase was modeled by using the correlation proposed by Moraga [90]. The lift force pushes the bubbles towards the wall. The lift coefficient in this model is calculated as

$$C_{\text{lift}} = \begin{cases} 0.00767 & \varphi > 6000 \\ -\left(0.12 - 0.2e^{-\frac{\varphi}{3600}}\right)e^{\frac{\varphi}{3} \times 10^{-7}} & 6000 \leq \varphi \leq 1.9 \times 10^5 \\ -0.002 & \varphi > 1.9 \times 10^5 \end{cases} \quad (\text{B-6})$$

where $\varphi = \text{Re}_b \text{Re}_v$. The aerodynamics lift force the between bubble and liquid shear is characterized by Re_b and Re_v is denoted for the lateral force between bubbles and vortexes created by bubble wakes.

In ANSYS Fluent, turbulent dispersion force is modeled with a general correlation which is based on Simonin's study [91]. This force defines the diffusion of bubbles caused by the turbulent eddies in the liquid phase. Turbulent dispersion force is defined as

$$\vec{F}_{\text{TD}} = -C_{\text{TD}} \rho_l k \nabla \alpha, \quad (\text{B-7})$$

where C_{TD} is the turbulence dispersion coefficient and k is the turbulent kinetic energy. The default value for turbulent dispersion coefficient is 1 in Fluent. The default virtual mass implemented into Eulerian-Eulerian multiphase model was employed in this study.

Energy or Enthalpy Equation

Conservation of energy/enthalpy for phase k in Eulerian-Eulerian multiphase flow model is described as follows.

$$\begin{aligned} \frac{\partial}{\partial t} (\rho_k \alpha_k h_k) + \nabla \cdot (\rho_k \alpha_k \vec{v}_k h_k) = & -\alpha_k \left(\frac{\partial P_k}{\partial t} + \vec{v}_k \cdot \nabla P_k \right) + \\ & \bar{S}_k : \nabla \cdot \vec{v}_k + R_k - \nabla \cdot \vec{q}_k + \sum_{k=1}^n \gamma_{ki} (T_i - T_k) + \dot{m}_{ki} (h_k - h_{ik}) \end{aligned} \quad (\text{B-8})$$

where γ_{ki} is the interphase heat transfer coefficient which is defined as

$$\gamma_{ki} = \frac{6\kappa_k \rho_i Nu_i}{d_i^2} \quad (\text{B-9})$$

In this equation, Nusselt number (Nu) is generally defined using the Ranz-Marshall correlation [92] which is given as

$$Nu_i = 2 + 0.6Re^{0.5}Pr^{0.3}. \quad (B-10)$$

In this correlation, Reynold number (Re) and Prandtl number (Pr) are calculated as

$$Re = \frac{\alpha_k \rho_k |\vec{V}_R| d_i}{\mu_k}, \quad (B-11)$$

and

$$Pr = \frac{C_{p,k} \mu_k}{\kappa_k}. \quad (B-12)$$

In Equations (B-11) and (B-12), $\vec{V}_R$ is the relative velocity between the phases, $C_{p,k}$ is the thermal conductivity of phase k and κ_k is packed bed conductivity expression. In Equation (B-8), $\vec{q}_k$ is the heat conduction term which is written as

$$\vec{q}_k = \alpha_k \kappa_k \nabla T_k. \quad (B-13)$$

VITA

Dündar Uçar

Dündar Uçar was born in Ankara, TURKEY, on 5 December 1979, the son of Bahtiyar Uçar and Halim Uçar. After completing his work at İnönü High School, Ankara, he went on to the Hacettepe University where he studied nuclear engineering and received his Bachelor of Science in May 2005. For the next two years he pursued a career as a research and teaching assistant at the Hacettepe University. He moved to State College, PA in August 2007. He entered the Graduate School at the Pennsylvania State University and started working at the Radiation Science and Engineering Center directed by Professor Kenan Ünlü. He also served as a graduate assistant in Mechanical and Nuclear Engineering Department and received his Master of Science in May 2009. He completed his PhD titled “Modeling and design of a new core-moderator assembly and neutron beam ports for the Penn State Breazeale Nuclear Reactor (PSBR)” in May 2013. Dr. Uçar contributed to international meeting and to be trained as a nuclear scientist and engineer in the United States. Currently, his main topics of interest are code development for nuclear applications, radiation transport applications including reactor physics, shielding, radiation detection and measurement, and several stochastic and deterministic calculation tools for design and analysis of complex systems as well as thermal-hydraulics analysis of nuclear systems using advanced codes.