Neutronic Design and Analyses of A New Core-Moderator Assembly and Neutron Beam Ports for The Penn State Breazeale Reactor

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Service Provided: Penn State Breazeale Reactor, Neutron Beam Laboratory

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Introduction

The Penn State Breazeale Reactor (PSBR), as a part of Radiation Science and Engineering Center (RSEC), was built in 1955 as a research and education hub. It is currently the longest operating research reactor in the United States. The initial reactor design used plate-type materials testing reactor (MTR) fuel elements with a 61-cm active fuel length and up to 93% uranium enrichment. Seven beam ports were built into the facility design for analyzing the nuclear properties of materials, determining reactor dynamics, and examining the effects of radiation on materials. After ten years of service, the reactor core design was changed to a TRIGA Mark III. The design conversion to a TRIGA core produced three major advantages for the reactor: (1) the reactor power was increased from 200 kW to 1 MW; (2) the reactor used fuel in the low-enriched safeguards category since TRIGAs use fuel that is 20% enriched in uranium, and (3) pulsing capability was added to the core due to the inherent prompt negative feedback characteristics of the TRIGA fuel elements, which are a matrix of uranium and $ZrH_{1.6}$ moderators. Unfortunately, the design conversion also resulted in a partial loss of experimental capability for the facility, such that use of six of the seven beam ports became limited. This is mainly due to the physical differences between MTR and TRIGA fuel element designs. Since the active length of a TRIGA fuel element (38.1 cm) is considerably smaller than the active length of an MTR fuel element (~61 cm), six beam ports, which were aligned with the MTR fuel, are now directed 12.7 or 27.9 cm below the core center. In this existing beam port configuration, only beam port (BP) 4 is located at the core center. In addition, five of the seven existing beam ports could not be properly aligned to the core-moderator assembly after the design change. A schematic drawing of the existing reactor core, D_2O tank, graphite reflector, and seven beam ports extended toward the reactor core are given in Figure 1. Therefore, the PSBR is not capable of simultaneously using all the available beam ports with the current configuration of the beam ports and the core-moderator assembly. Only two beam ports, namely BP4 and BP7, are coupled with the reactor core for experimental purposes. BP7 is mainly used for neutron transmission measurements. This port is 12.7 cm below the core center and thus the neutron flux at this beam port is significantly lower. Almost all of the other experimental techniques, i.e. neutron imaging, neutron depth profiling, detector testing and development etc., are conducted at BP4. However, the high content of the prompt gamma-rays in these beam ports affects all of the experiments conducted in the facility. The prompt gamma-rays are produced by the neutron capture of the hydrogen in the pool's water due to the $^1H(n,\gamma)^2H$ reaction, which mainly takes place at the sides of the D_2O tank, see Figure 1.

This study presents a new PSBR core-moderator assembly design and five new beam ports, which would eliminate all the limitations of the existing design by increasing the number of simultaneously used beam ports from two to five and by mitigating the amount of prompt gamma-rays in the beam port facilities. The major constraints of the PSBR are mainly geometric factors such as available infrastructure in the beam hall, the tower design, geometrical arrangement of the beam ports, and the core and moderator designs.

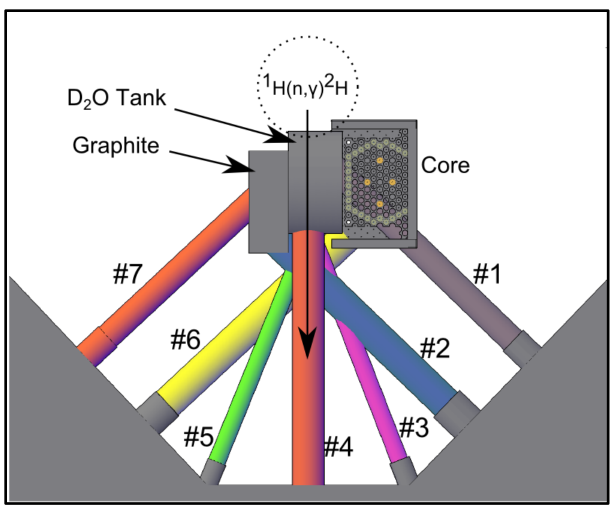

FIGURE 1: A schematic drawing of the PSBR core-moderator assembly layout with the graphite reflector and the beam ports extended to the reactor core.

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Furthermore, the prompt gamma-ray contamination problem and thermal-hydraulics safety of the core are design parameters and the neutronic performance of the proposed design are calculated by detailed neutronic simulations and discussed below.

Design Considerations

The existing core-moderator assembly design is the main cause of the geometric mismatch of the beam port configuration. The key parameter in the design process is the calculation of the optimal size and shape of the moderator tank. The limitations of the PSBR were previously studied by Butler, who specifically analyzed the utilization of three moderator tank shapes for the PSBR (crescent, horseshoe and rectangular) and the geometric arrangement of five new neutron beam ports in a moderator tank [1]. In this study, a crescent-shaped moderator tank is favored since it allows for the simultaneous use of five beam ports. After the selection of the moderator tank shape, the second design step is the proper coupling of the moderator tank with the reactor core in order to eliminate the prompt gamma-ray contamination problem by minimizing pool water at the interface of the core-moderator assembly. This was achieved by keeping the faces of the top and bottom grid plates and the crescent-shaped moderator tank as close as possible (0.62 cm between the core and the moderator tank). The final step in the design process is how to support a new core design with a new reactor tower. The existing reactor core is supported by a tower through the bottom grid plate. The top grid plate is connected to the bottom grid plate. In the new design, the top and bottom grid plates are equal in size and smaller than the existing grid plates. As a result, the tower design will be changed by installing four new support bars and two supports plates on top of the core. Figure 2 shows the core-moderator assembly and tower design for the PSBR after the design changes.

For the redesigned reactor, four thermal and one cold neutron beam ports are proposed for various neutron techniques. Since the cold neutron beam port will channel three curved neutron guide tubes, seven instruments can be simultaneously used in the beam hall. Four new techniques are planned for the facility: Triple-Axis Spectrometry, Prompt Gamma Activation Analysis (PGAA), Convectional and Time-of-Flight (TOF)-Neutron Depth Profiling (NDP), and Neutron Powder Diffraction (NPD). In addition, the existing neutron transmission and neutron imaging facilities will remain available.

Optimization and Performance Analyses of New Core-Moderator Assembly Using Simulations

Design and neutronic simulations of the new reactor core were performed with a reference core model selected as loading 53H, which went critical in May 2009. There are 102 fuel elements, ten graphite rods, two dry (air-filled) tubes, three fuel-follower control rods (shims, regulator, and safety) and one air-follower control (transient) rod. The ten graphite rods at the periphery of the core were removed to achieve proper coupling with the crescent-shaped moderator tank. The main function of the graphite rods is to enhance neutron economy as well as to decrease the critical uranium mass required to achieve criticality. Therefore, a decrease in the excess reactivity of the reactor after removal of these rods is expected. On the other hand, the new moderator tank is bigger in size and covers more than half of the core periphery, which will result in a positive impact on the excess reactivity of the system. These two competing design changes on the excess reactivity of the reactor were analyzed.

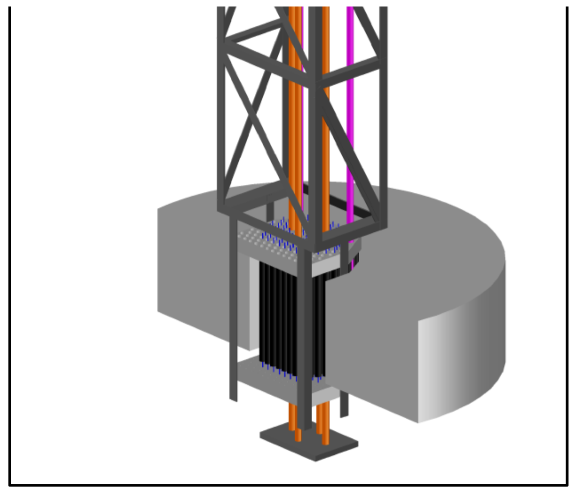

FIGURE 2: 3D CAD drawing of the new core-moderator assembly and tower design.

The second part of the neutronic analysis is the calculation of the optimal configuration of the core-moderator assembly and neutron beam ports, which will provide high-flux thermal neutron beams with minimal background radiation to the beam port facilities. The simulation was performed using Monte Carlo N-Particle ver. 5 (MCNP5) code with sensitivity studies [2]. First, the design parameters were evaluated. The simulation approach was to make a sensitivity study by changing a parameter of interest, such as moderator size, while keeping all the other parameters unchanged. The effect of the design parameter on the neutronic performance of the new beam ports was examined in successive MCNP calculations. The optimal value of the design parameter of interest was selected to yield the maximum thermal to fast neutron flux ratio and minimum gamma-ray dose at the exit of the new beam ports.

After the optimal design parameter values were calculated, the neutronic performance of the new reactor with five new beam port designs was analyzed by comparing the neutron and gamma-ray flux distributions to the measured and simulated spectra in the existing BP4 and BP7. The filter and collimator systems in each new beam port were selected to accommodate the requirements of the neutron beam technique for the beam port of interest.

Reactivity Analysis

The excess reactivity of the new reactor core was analyzed by using MCNP5 and TRIGSIMS (TRIGA Simulator-S), the fuel management code system of the PSBR based on Monte Carlo methods [3]. TRIGSIMS provides a user interface to define the reactor geometry and calculate the depleted fuel element compositions by coupling the ADMARC-H nodal diffusion code, the MCNP code, and the ORIGEN-S depletion code. TRIGSIMS estimates the effective multiplication factor of the system, the axial power distribution for five axial nodes in each fuel element from the MCNP output, and the depleted fuel composition of each element from the ORIGEN-S output. It is also capable of calculating the reactor core excess reactivity and the integral reactivity worth of the control rods. In the MCNP calculations, TRIGSIMS employs a temperature and burnup dependent cross-section data set generated by the NJOY (v99.0) code [4] for 20 selected important isotopes and predictor-corrector methodology is used for the depletion scheme. ADMARC-H is used to provide the initial source distribution for the MCNP calculation. This avoids using large number of inactive cycles in the MCNP calculation.

In the reactivity analysis, two MCNP models were analyzed and the effective multiplication factor of the reactor was calculated in each model. The first model contains only the core and the second one has the core with the moderator tank in the pool. As a result, any increase observed in the multiplication factor is due to the presence of the moderator tank. The radius of the moderator tank was set to 60 cm around the central thimble (core center) in the second model. All the control rods were modeled at their critical rod positions for 1-MW power operation using data obtained from the reactor operations staff. Then, the contribution of the crescent-shaped moderator tank to the core excess reactivity was calculated by using the following relation:

$$\text{excess reactivity (\$)} = \frac{\mathrm{k_{eff1}} - \mathrm{k_{eff2}}}{\mathrm{k_{eff1}}\mathrm{k_{eff2}}\beta_{\mathrm{eff}}} \qquad (1)$$

In the preceding equation, k_{eff1} is the effective multiplication factor of the core-moderator assembly, k_{eff2} is the effective multiplication factor of the bare core and β_{eff} is the effective delayed neutron fraction, which is 0.007 for TRIGA reactors. As a result of this calculation, the contribution of the new moderator tank to the excess reactivity of the system was estimated at \$1.13 with a standard deviation of 6×10^{-5}. This is much higher than the excess reactivity contribution of the existing D_2O tank, which was measured as \$0.68. By taking into account the decrease in the excess reactivity of the system due to the removal of the graphite rods, the existing D_2O tank and the graphite block, it is expected that the new core-moderator assembly design will provide \$0.15 higher excess reactivity to the reactor.

Evaluation of Optimal Design Parameters

The goal for the project is the determination of the optimal dimensions of the moderator tank, beam ports, re-entry holes and collimators to enable maximal neutron output with minimal background radiation in the beam port facilities. However, some possible values of the design parameters are restricted by extrinsic constraints and are not taken into account in this analysis. For example, the height of the moderator tank was set to 50.8 cm about the fuel element centerline in order to provide 5-cm openings between the grid plates and the moderator tank. This was intended to increase the cooling water cross-flow rate in the core, which is significant to the cooling of the PSBR fuel [5]. The relative tilt angles between the new beam ports (NBPs) and the core are restricted by the location of the experimental facilities in the beam hall. Moreover, the beam ports are directed to the center of the fuel elements since the axial neutron flux profile is a cosine with its maximum at the center of fuel elements. The two design parameters identified in the optimization study were the radius of the crescent-shaped moderator tank and the distance between the core front faces and each beam port in the moderator tank. The optimal value of a design parameter of interest was determined by making successive MCNP simulations while keeping all the other parameters unchanged and then estimating neutron and gamma-ray flux spectra at the end of each new beam port. As a result, the effect of varying the design parameters on neutronic performance of the new beam ports was estimated.

The distance between the reactor core and the end of each new beam port would not permit high precision in the MCNP results due to excessive computational time requirements. This was overcome by dividing the problem domain into two parts and running the MCNP simulations in two steps. Following that, non-analog Monte Carlo methods or variance reduction techniques were applied to decrease the uncertainty in the simulation results. In the first part, a criticality calculation was performed for a full core model that created a neutron and gamma-ray source on a virtual surface defined at a depth of ~1.2 meters into each new beam port. Then, a separate fixed source calculation was performed in each individual beam port model starting from the new virtual surface source. The angular distribution of neutrons and gamma-rays on the source surface was modeled by using a conical source direction-biasing technique. This technique restricts the source emission to a set of nested cones, while sampling the neutrons on the source surface in the beam port models. The particle direction and probability of emission on the surface source were defined by using source probability and source bias cards in the MCNP input. The calculated spatial distributions of the source particles in the new beam ports were almost uniform. The flux tallies at the end of each NBP were estimated by a point detector tally, which gives deterministic estimates of particle flux at a point in space. Point detector estimation is mainly employed to force particles to a region where it is very difficult or impossible to naturally transport the particles in the MCNP calculations.

The MCNP simulation methodology was first verified with the measured neutron flux, which is $3.0\text{x}10^7 \pm 0.3\text{x}10^7 \frac{\text{n}}{\text{cm}^2.\text{s}}$, with a single-disk-slow-chopper time-of-flight by Niederhaus at the exit of the existing BP4 [6]. The flux tallies were normalized to the number of starting source particles in the MCNP outputs and corrected by proper scaling factors to obtain the 1-MW power operation tally results. For the gamma-ray flux, the scaling factor was estimated by using the measured gamma intensity (2460 γ/s ± 25) at the exit surface of BP4.

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

1. The depleted fuel element compositions in core loading 53H that were estimated by the TRIGSIMS code at the beginning of the cycle were employed.
2. Control rods were modeled in an "all rods out" state in which the B_4C absorber section is totally out of the fuel element active region.
3. The temperature of the fuel section in all of the elements and in the water within the fuel elements was set to 800 K and 300 K, respectively, since we are still developing a suitable thermal-hydraulics model to predict the temperature distribution in the core.

As a result of this calculation, the total thermal neutron flux (<0.55 eV) at the exit of BP4 was estimated as $2.86\text{x}10^7 \pm 0.0345\text{x}10^7 \frac{\text{n}}{\text{cm}^2-\text{s}}$, which agrees well with Niederhaus' measurement. In addition, the MCNP-estimated thermal neutron flux spectra at the end of the existing BP4 match very well with the measured and theoretical Maxwell-Boltzmann distributions as shown in Figure (3). These results clearly indicate the success of the MCNP simulation approach, and thus it was applied to all the MCNP calculations throughout the study.

Optimal Distances between the Core and the New Beam Ports

After the verification of the MCNP simulation approach, it was applied first to the determination of the optimal distances between the new beam ports and the core faces. The beam ports were directed to the core center and the relative orientation with the determined tilt angles were included in the MCNP models (see Figure 4). The input parameters and the assumptions given in the previous section were applied to these calculations with the exception that all the fuel elements were modeled with fresh fuel compositions for ease in computation. About 300 million particles were simulated in the full core and beam ports models in order to decrease the relative error in the tally results below the acceptable level, which is considered to be 1% for flux tallies and 5% for point detector tallies. The MCNP simulations were performed in parallel mode in a high performance cluster with 64 processors. For additional simplicity, new beam ports were modeled as empty tubes with inner diameters of 15.24 cm (6 in). For the initial case, the moderator radius was set to 76.2 cm (30 in). In order to take advantage of the inherent two-fold symmetry in the core, only NBPs 2, 3 and 4 were analyzed.

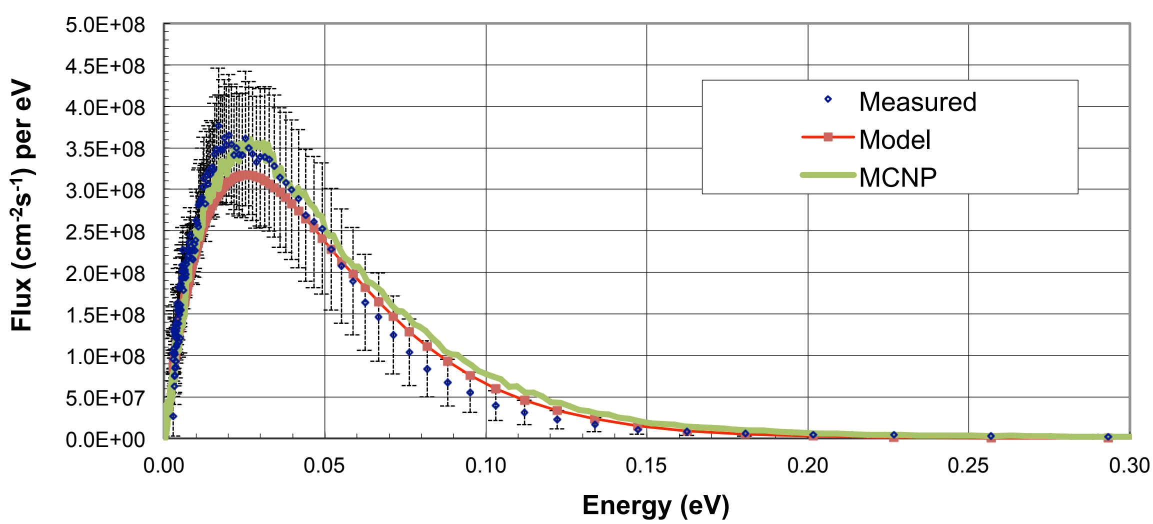

FIGURE 3: Comparison of the MCNP-calculated thermal neutron flux at the end of existing BP4 to the measured [6] and theoretical Maxwell-Boltzmann distribution

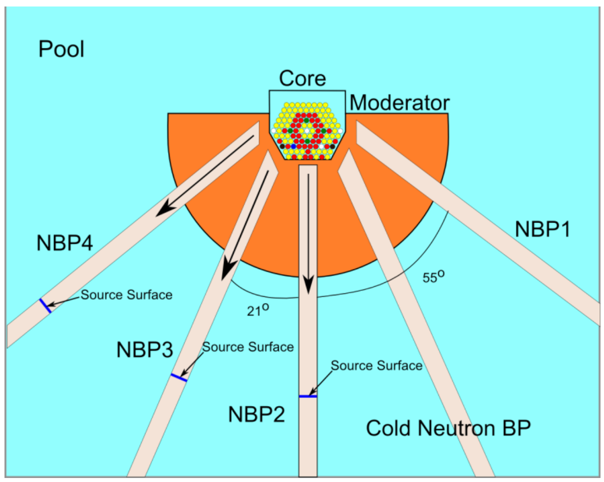

FIGURE 4: MCNP model employed in the calculation of optimal beam port locations.

The calculation methodology is summarized as follows. The distance between each new beam port and the core is increased, the simulation is performed and the neutron and gamma-ray flux distributions are estimated at the end of the beam port with the two-step calculation. The relative tilt angles of the beam port shown in Figure 4 are relatively fixed considering the available infrastructure and the locations of the new instruments in the beam hall. The locations of the surface source employed in the calculations are also shown in the figure. The resulting thermal and fast neutron flux estimated as a function of distance between the new beam ports and the core are given in Figures 5, 6, and 7. The results indicate that as the start of the beam ports retreat from the core face, the thermalization volume increases in each beam port. Accordingly, the fast neutron flux decreases and the thermal neutron flux increases in the beam ports. However, after a specific distance, the proportion of thermal neutrons scattered out of the path by the heavy water becomes higher than the proportion entering the beam port. That is why a considerable decrease in the thermal neutron flux was observed in the new beam ports as their distances from the core increased. Therefore, there exists an optimal separation distance that maximizes the thermal neutron flux at the end of the beam port that differs for each NBP. The effect of the thermalization volume is more significant for NBP3 and NBP4, which are located close to the periphery of the moderator tank, compared to NBP2. This effect can be seen in Figures 5–7.

The total gamma-ray flux as a function of distance between the core and the entrance to NBP2 is shown in Figure 8. It can be seen that the total gamma-ray flux decreases as the distance increases. Similar results are obtained for the other beam ports. This is a secondary optimization parameter to the thermal neutron flux.

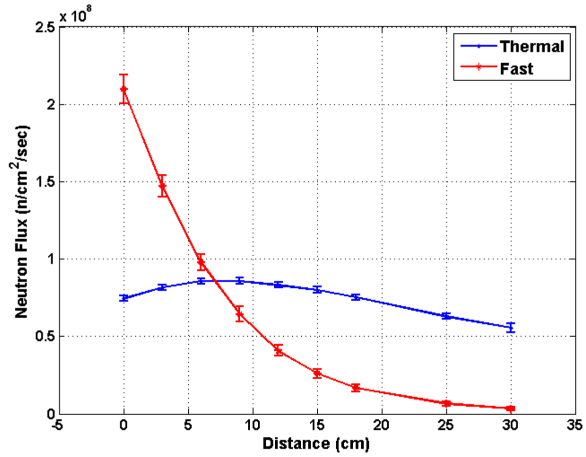

FIGURE 5: Total thermal/fast neutron flux as a function of distance between the core and NBP2.

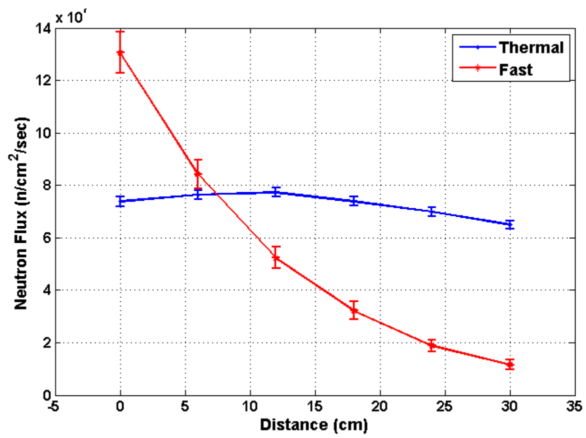

FIGURE 6: Total thermal/fast neutron flux as a function of distance between the core and NBP3.

In order to provide the highest thermal neutron flux to the experimental facilities, the optimal distances between the core's outer surfaces and NBP1, Cold Neutron Beam Port, NBP2, NBP3, and NBP4 were selected as 12 cm, 18 cm, 15 cm, 18 cm and 12 cm, respectively. At these selected distances, the total thermal neutron flux is slightly less than the maximum value but the total fast neutron and total gamma-ray fluxes are significantly reduced.

Determination of the Optimal Size of the Moderator Tank

The second part of the optimization study is the determination of the optimal radius of the moderator tank. Its height is set to 50.8 cm to increase the essential cooling water cross-flow entering into the core. Similar to the previous sensitivity study, the total thermal-to-fast neutron ratio and the total prompt gamma-ray flux at the exit of each new beam port was studied as a function of moderator tank radius in successive MCNP calculations with a two-step

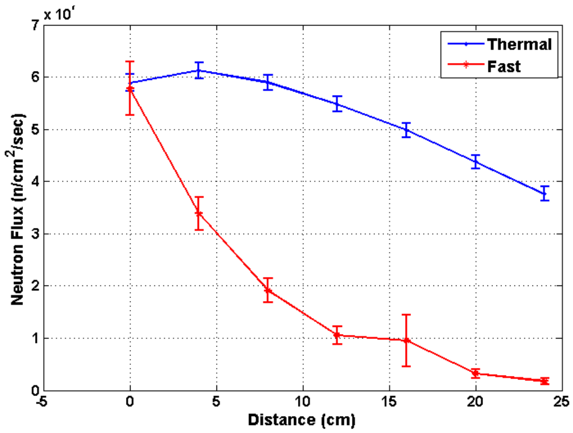

FIGURE 7: Total thermal/fast neutron flux as a function of distance between the core and NBP4.

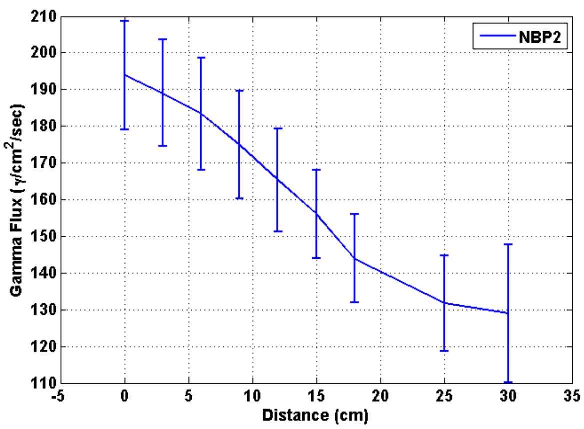

FIGURE 8: Total prompt gamma-ray flux as a function of distance between the core and NBP2.

approach. In this case, the new beam ports were modeled at their optimal distances calculated in the previous section and the total tank radius was varied.

Figure 9 shows the MCNP-predicted thermal-to-fast neutron ratio at the end of NBP2 as a function the moderator tank radius. As seen from this result, putting more D_2O around the beam port does not increase the thermal flux after the moderator tank radius becomes 76.2 cm (30 in). This is only true for NBP2; the other beam ports are located close to the periphery of the moderator tank so that the thermal-to-fast neutron ratio continues to rise as the tank size increases. However, the limited amount of heavy water available restricts the moderator tank radius to 76.2 cm. A considerable gamma-ray flux increase can be observed in the NBPs as the tank size increases. Therefore, the moderator tank radius was selected as 76.2 cm (30 in) around the core center (center thimble).

Design and Analysis of New Beam Ports for the New PSBR

The neutronic performance of the new beam ports is not only affected by the core-moderator assembly

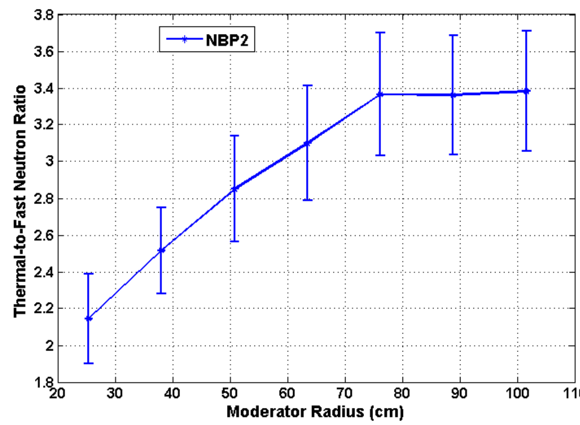

FIGURE 9: Thermal-to-fast neutron ratio at the end of NBP2 as a function of moderator tank radius.

design but also the beam divergence, collimator system, filter material and other geometric factors like physical dimensions. In the optimization study, the neutronic design of the new reactor was explored with five beam port models without considering these factors. However, the final design features of each neutron beam port will be based on the experimental facility to be used. The lowest estimated thermal and cold neutron flux for the selected facilities with the beam port locations are given in Table 1 [1]. Detailed design features of each new beam port are given below.

The geometry, filter material, and collimator system that will be employed in NBP1 and NBP2 are based on the NDP beam port designed by Ünlü at the University of Texas (UT) research reactor [7]. The NDP beam port at the UT research reactor has two steel tube sections with 15.4 cm and 20.6 cm outer diameters. A single-crystal hemlite-grade sapphire filter, which is 17.2 cm long and 3.56 cm in diameter, is located at the end of the first tube. The collimator system is located in the second tube and its annuli are made of several steel and lead sections with several sheets of BORALTM between the steel sections. The same filter material and collimator system will be installed in NBP1 and NBP2 with a similar geometric configuration. However, the length of the sapphire filter will decrease to 7.62 cm and the lengths of the steel tubes in these beam ports will differ.

The NBP3 design will have the same geometric parameters as the transmission beam port (BP7) in the existing facility. This port has four sections and three empty tubes in which the neutrons are channeled to the measurement facility. In addition, a BF_3 detector is located at the end of one of the neutron channels to continuously monitor the variations in the neutron flux in the facility. For NBP4, the same filter and collimator system designs employed in the existing BP4 will be used. BP4 has a bismuth filter, lead, and borated aluminum in the first section of the beam port close to the reactor core and a collimator system of annuli

TABLE 1: Summary of techniques in the redesigned PSBR along with the lowest estimated neutron fluxes of similar instruments at other research reactors [1].

Location	Technique	Lowest Utilized Flux ($n/cm^2/sec$)
NBP1	Triple-Axis Spectrometry	1.72×10^5 (thermal) 2×10^5 (cold)
NBP2	Free Beam (Explanatory)	1×10^6 (thermal)
NBP3	Neutron Transmission	1×10^4 (thermal)
NBP4	Neutron Imaging	3×10^6 (thermal)
GT1*	Prompt Gamma Activation Analysis	2.4×10^7 (thermal) 5x107 (cold)
GT2*	Neutron Powder Diffraction	1×10^5 (thermal)
GT3*	Conventional and Time-of-Flight Neutron Depth Profiling	1.5×10^8 (thermal) 2.5×10^9 (cold)

*Part of the cold source BP. GT = Guide Tube

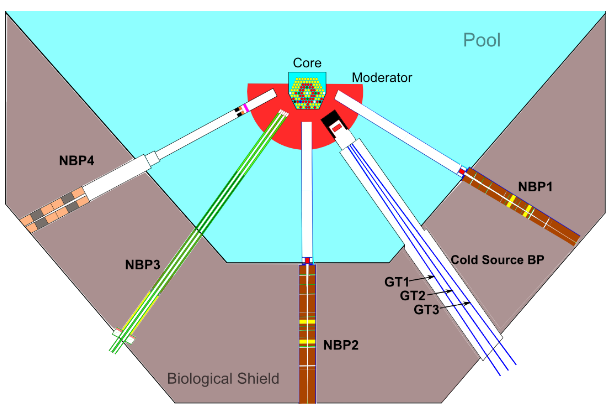

FIGURE 10: The design features of the new beam ports in the proposed core-moderator assembly.

made of several concrete and lead sections located at the end of the beam port. The design features of the new beam ports with the new core-moderator assembly are shown in Figure 10.

Similar to NBP1 and NBP2, the cold neutron beam port in the new facility has been selected based on the University of Texas Cold Neutron Source (UTCNS) [8]. This beam port contains an aluminum moderator chamber, a cooled mesitylene moderator, a cold source cryostat system and a neutron guide system. The moderator chamber is a 7.6-cm diameter and 2-cm thick cylinder and the cryostat system consists of a Two-Phase Closed Thermosyphon with a Reservoir (TPCTR), a helium cryorefrigerator and a vacuum box. All the neutron guide sections provide a 300-m radius of curvature and contain three vertical channels. The guide tube walls are coated with a ^{58}Ni layer with a 1000-Angstrom thickness. The PSBR cold neutron beam port will have a similar moderator chamber, mesitylene moderator and TPCTR cooling system with a better operating range and heat removal capacity in the cryostat system [9]. Three guide tubes will be available to utilize the cold neutrons in the cold neutron beam facilities. The coating material will be replaced by a super-mirror that had not been selected at the time of this study.

Neutronic Performance Evaluation of the New Beam Ports

The neutronic performance of each new beam port was evaluated by comparing the MCNP-estimated neutron and gamma-ray flux spectra to the measured and MCNP-estimated neutron spectra in existing BP4 and BP7. The previous MCNP simulation methodology, performed in two steps with full core and beam port models and conical source direction biasing and point detector estimate variance techniques, was followed in this study. Instead of the TRIGSIMS code, the depleted fuel element compositions were calculated by the Burned Coupled MCNP Simulation tool developed for the PSBR core [10] with new cross-section data generated using the NJOY-99.264 code from ENDF/B-VII data tapes [11] for 86 isotopes with 10 K temperature increments between 293.72 K and 900 K. The accurate modeling of the TRIGA fuel is important since the up-scattering probability in the ZrH moderator increases with the temperature and it adversely affects the neutron economy in the reactor. The fuel temperatures in all of the fuel elements and control rods were estimated by making a linear interpolation between the measured fuel temperatures in instrument rod 16 (I-16) and the estimated local power generation in each fuel section using the Burned Coupled MCNP Simulation tool.

Because of the lack of the available cross-section data in the evaluated data tables, the neutron transmission rate in the 7.62-cm thick hemlite-grade sapphire filter was estimated by employing Cassels function [12], which defines the attenuation cross-section (in barns) as:

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

where λ is the neutron wavelength. The fitting parameters used in this equation were given by Mildner et al. [12]. Although a two-step calculation approach was followed in NBP1 and NBP2, an additional calculation step was performed to calculate the neutron transmission in the sapphire filter. In the full core model, the neutrons were tallied on the sapphire filter entrance surface. Then, the thermal neutron transmission rates in the sapphire filter were calculated with the use of Cassels function as 74.91% ± 3.56% for NBP1 and 73.32% ± 2.43% for NBP2. The fast neutron attenuation rates were estimated as 78.15% ± 5.60% in both beam ports. By ignoring the small angle scattering of the transmitted neutron beam after the sapphire filter, it was assumed that the angular distribution of the neutron beam on the entrance surface would not change. The neutrons were sampled starting from the exit surface of the sapphire filter in the beam port model. The estimated fast neutron attenuation rate in the sapphire filter is less than the expected value (90%), likely because of the reported discrepancies of Cassels function in the fast energy region [12].

The cold neutron beam port was modeled up to the exit of biological shield and the ^{58}Ni coating material was used on the guide tubes in the MCNP calculations. Additionally, the calculated cold neutron flux at the end of each guide tube was converted to the thermal equivalent neutron flux and used in the performance evaluation. The total thermal and fast neutron fluxes and the gamma-ray flux were determined at the exit surface of the NBPs and guide tubes and then compared with the MCNP simulation results in existing BP4. The estimated neutronic performances of the NBPs with respect to BP4 are tabulated in Table 2. For the guide tubes, the gamma-ray flux was not considered in the performance evaluation since they will have a 300-m radius of curvature in the actual design to eliminate the mono-directional prompt gamma-rays from the scattered neutron beam. NBP3 provided the same neutronic performance as the existing BP7 except its gamma-ray component was significantly reduced. As seen from the calculated results, the best neutronic performance improvement was attained in NBP2 (as expected) because it is directed along the core centerline. The new core-moderator assembly design eliminated the prompt gamma-ray contamination problem in the facility, such that the gamma-ray flux decreased by a factor of more than 100 in the NBPs compared to the existing BP4. The smallest improvement in the NBPs performance was observed in NBP4, which has the same design features with existing BP4. Although it is almost symmetrically installed to the edge of the moderator tank similar to NBP1, the fast neutron component of NBP4 is considerably higher and its thermal neutron beam component is lower. This is mainly due to the use of the bismuth crystal as the filter material in this beam port. Although bismuth is very effective for filtering the gamma-rays, it decreases the thermal neutrons and is not an effective filter for the fast neutrons. On the other hand, NBP1 employs a hemlite-grade sapphire crystal, which is one of the best filter materials for the gamma-rays as well as fast neutrons [12]. A sapphire crystal can eliminate 50% of the gamma-rays and 90% of the fast neutrons, which is why it will be used in NBP4.

Summary and Conclusion

This study sought to eliminate inherent design problems of the PSBR while preserving its advanced features by implementing a new core-moderator assembly with five new neutron beam ports.

The major limiting components of the existing reactor design are identified as the core and moderator tank configuration. The dimensions of the moderator tank are not appropriate for the utilization of more than two beam ports in the facility. Moreover, the support plates of the suspension tower, which carry the reactor core, restrict the coupling of the core and moderator tank. Therefore, the coupling of a new moderator tank with the core is achieved by changing the shape and size of the top and bottom grid plates and the support structure on the tower. A crescent-shaped moderator tank is employed in the new design and proper coupling is attained by matching the shapes of the core and moderator tank at the interface. The NBPs are directed to the core center to minimize the number of hydrogen capture prompt gamma-rays in the beam port facilities. After all of the design changes, the inherent design problems of the existing reactor are eliminated while maintaining the ability to move the reactor to other experimental facilities.

TABLE 2: Neutronic performance of the NBPs compared to the existing BP4.

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	8.6E-3
NBP4	1.23	1.44	4.2E-3
GT1*	1.75	0.87	—
GT2*	2.05	1.18	—
GT3*	1.96	1.65	—

*Part of the cold source BP.

In the second phase of the study, the individual design of each new beam port, including filter material, collimation system, beam divergence, etc. is discussed. Five neutron beam port designs to be used for several thermal and cold neutron beam port facilities are introduced. The design features of NBPs are selected based on the experimental facilities to be used. The NDP beam port and UTCNS utilized in UT research reactor as well as BP4 and BP7 in the existing PSBR are implemented in the selected facilities and techniques.

Modeling of the redesigned PSBR was completed with highly detailed neutronic simulations using MCNP, TRIGSIMS, and the Burned Coupled MCNP Simulation Tool to identify the optimal design parameter values. The MCNP modeling methodology demonstrated in this study is very fast and successfully estimated the neutron and gamma-ray flux spectra with reasonable accuracy, as verified by agreement with experimental results. Furthermore, computer simulations demonstrated that the new design will provide significant performance improvements in the utilization of the thermal and cold neutron beams to the experimental facilities. The hydrogen prompt gamma-ray contamination problem, which is the main source of background radiation in the experiment, will be significantly diminished in the new beam ports. In conclusion, the new PSBR core-moderator assembly and beam port configuration will fulfill the facility's need for improved beam port facilities.

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