International Conference on Mathematics, Computational Methods & Reactor Physics (M&C 2009)
Saratoga Springs, New York, May 3-7, 2009, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2009)

SENSITIVITY STUDY TO OPTIMIZE THE THICKNESS OF A MODERATOR FOR USE IN THE AP1000 SOURCE, INTERMEDIATE, AND POWER RANGE EXCORE DETECTORS

Joel A. Kulesza
Westinghouse Electric Company, LLC
PO Box 158 Madison, PA 15663 USA
kuleszj@westinghouse.com

ABSTRACT

The purpose of this paper is to discuss the coupled deterministic-stochastic analysis techniques and codes used to optimize the thickness of a polyethylene moderator for use in the AP1000 excore source, intermediate, and power range detectors. The analysis incorporated a "bootstrapping" approach whereby an outward directed cylindrical boundary source was generated by DORT, source particle cumulative density functions (CDFs) in space, direction, and energy were formulated, these CDFs were processed by another code to create discrete source particles, and these discrete source particles were used by MCNP to calculate response functions which were meant to approximate detector signal responses. Furthermore, a simplistic stochastic study was performed to determine the maximum moderator thickness that should be considered for this application. For the AP1000, the optimum moderator thickness was the maximum allowed due to the mechanical design (1.38 cm) whereas for a general case the optimum moderator thickness is approximately 7.5 cm.

Key Words: Excore Detector, MCNP, Moderator, Optimization

1. INTRODUCTION

It was believed that excore detector signal strength could be increased by moderating some of the fast neutrons in the vicinity of the active volume of the detector to thermal energies in order to encourage a higher reaction rate in the detector volume to produce greater signal strength. The use of a polyethylene moderator had been successfully used previously at another domestic nuclear power plant to achieve this effect.

The purpose of this paper is to document efforts surrounding the calculation of the optimum thickness of a polyethylene moderator which surrounds the active detector volume of the AP1000 source, intermediate, and power range excore detectors. Both BF_3 and fission chamber detectors were considered. This calculation acts as a case study of how the author was able to use a novel modeling approach whereby a relatively fast-running deterministic calculation was used to generate a complex and reactor-specific fixed source for a highly-detailed Monte Carlo perturbation study.

2. CALCULATION PROCESS SUMMARY

Since it was believed that excore detector signal strength could be increased by moderating some of the fast neutrons in the vicinity of the active volume of the detector to thermal energies in order to encourage a higher reaction rate in the detector volume, a series of perturbed cases were set up and executed to examine this behavior by considering varying moderator thicknesses. Historically, another domestic nuclear power plant used a solid polyethylene moderator to achieve this effect so this material was chosen as the moderator for this application as well.

In order to model this system, a coupled deterministic-stochastic radiation transport analysis was performed. A representative two-dimensional r-z discrete ordinates transport model was used to generate an outward-facing cylindrical shell neutron boundary source which was post-processed to determine a discrete source neutron distribution with particle-specific initial positions, directions, and energies. This discrete neutron distribution was incorporated into a series of Monte Carlo transport joint neutron-photon models using the MCNP code[1] to determine the relative effect of different moderator thicknesses on neutron reaction rates in the active detector volume at the source, intermediate, and power range excore detector locations. The benefit of using a representative deterministic model is the capability of accurately modeling the source neutron distribution in space, energy, and direction as well as acquiring accurate normalization data to predict detector reaction rates.

The two reactions considered were ^{10}B(n,α) and ^{235}U(n,f) to represent BF_3 and fission chamber detectors, respectively. Two 47-group neutron interaction cross-section sets, based on the BUGLE-96 group structure[4], were calculated for these two reactions. Both cross-section sets are shown in Section 2.4. These cross-sections were used as response functions in MCNP to generate volumetric flux tally values for each detector for each reaction (leading to six tallies per moderator thickness considered). The range of moderator thicknesses considered was from no moderator to a maximum thickness based on the detector dimension specifications provided by the manufacturer, described in Section 2.5.1.

Furthermore, a simplistic study of a wider range of moderator thicknesses was performed. Using MCNP's internal SDEF functionality, a point, mono-directional (beam), fission spectrum neutron source was set incident upon a slab of polyethylene of varying thicknesses from 0 cm to 20 cm. While not truly representative of the AP1000 geometry and flux spectrum, this should allow "order of magnitude" estimates of the effect of moderator thicknesses outside those explicitly modeled for the AP1000. This proved useful in verifying the behavior in the preceding analysis as well as confirming the overall behavior of a polyethylene moderator and determining where the uninfluenced peak thickness exists.

2.1. Discrete Ordinates Geometry Generation

A representative model of the AP1000 reactor cavity and surrounding geometry for the GGDM code already existed and was therefore used. GGDM is primarily a meshing code used to create two-dimensional discrete ordinates input from combinatoric geometry and is distributed with the BOT3P suite of codes[2] through the ENEA. The geometry was primarily manipulated in GGDM and processed to create an r-z DORT mesh because of the ease associated with specifying and modifying combinatoric geometry. This mesh was integrated into a DORT input deck using appropriate problem definition cards and is shown in Fig. 1.

The DORT code is an industry standard multiple geometry two-dimensional discrete ordinates transport code also distributed by ORNL in the DOORS code package[3]. In this analysis,

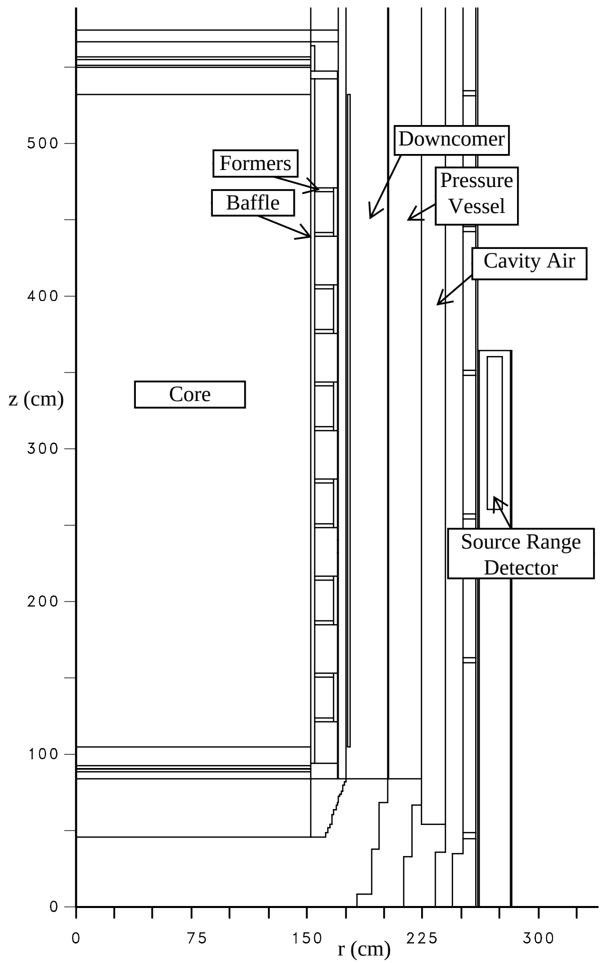

Figure 1. Axial view of the AP1000 reactor cavity geometry as modeled in DORT.

anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. The fixed distributed source used was generated by a proprietary code that considers beginning and end of cycle burnups, assembly enrichments, pin-by-pin power distributions, and core power density to produce mesh-wise source values. The cross-section set used was the standard BUGLE-96 set where that is a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application[4]. The execution of the r-z DORT case yielded an outward directed cylindrical boundary source file that described the angular flux at the outside surface of the pressure vessel.

2.2. Sampling the Boundary Source File

DOMINO is a code distributed by the NEA, and previously RSICC, that reads DORT-generated internal boundary source information and writes an output file containing appropriate cumulative probability density functions (CDFs) that can be used to produce source particle definitions. As such, a DOMINO input deck and supporting files were created to generate a cylindrical shell surface source with radius corresponding to the outer radius of the pressure vessel (R_s) and axial extents from the bottom (Z_0) to the top (Z_{max}) of the active fuel length. An assumption was made that neutron leakage through the pressure vessel between these two axial extents was significantly higher than leakage above or below the active fuel region.

As DOMINO is executed, it creates a series of normalized CDFs that describe azimuthal (m), polar (l), spatial (i), and energy (g) distribution spectra, or

$$E_{gilm} = \frac{X}{\sum\limits_m X}; D_{gil} = \frac{\sum\limits_m X}{\sum\limits_l \sum\limits_m X}; C_{gi} = \frac{\sum\limits_l \sum\limits_m X \Delta S_i}{\sum\limits_i \sum\limits_l \sum\limits_m X \Delta S_i}; B_g = \frac{\sum\limits_i \sum\limits_l \sum\limits_m X \Delta S_i}{\sum\limits_g \sum\limits_i \sum\limits_l \sum\limits_m X \Delta S_i}, \tag{1}$$

where $X = \Psi_{gilm} \left| \mu_{lm} \right| W_{lm}$ and $\Delta S_i = 2\pi R_s \left(Z_{i+1} - Z_i \right)$ for a cylindrical surface.

For the Monte Carlo execution, it is important to note that the denominator of B_g is an overall source particle normalization factor,

$$A = \sum_g \sum_i \sum_l \sum_m \Psi_{gilm} \left| \mu_{lm} \right| W_{lm} 2\pi R_s \left(Z_{i+1} - Z_i \right). \tag{2}$$

This value represents the total number of particles crossing the boundary surface summed over all energies, space, polar, and azimuthal angles at a particular radius. Therefore, A is a conversion factor from a MCNP per-particle normalized flux to an absolute flux.

2.3. Discrete Source Particle Generation

MCNPBQ is a Westinghouse-internal code that randomly samples DOMINO-generated CDF files to create batches of source particle initial positions, directions, and energy data catalogued in a source particle file (SPF). The source particle location is determined by randomly sampling the spatially dependent CDFs produced by DOMINO to define the Cartesian coordinates x, y, and z of the source particle. In addition, the initial direction of the source particle is randomly sampled from the angular dependent CDFs (which are defined by the angular quadrature of the DORT calculation) and the direction cosines of the source particle, u, v, and w are defined. Finally, the particle's initial energy is determined by sampling the energy dependent CDF and the particle is assigned the group average energy as defined in the BUGLE-96 energy spectrum.

In this case, a total of five million source particles were generated of which only one million were required (with the excess kept in case a more detailed study was required later). It should be noted that the azimuthal dependence of the reactor cavity flux could not be explicitly modeled due to a limitation of the DOMINO code that only allows it to read r-z internal boundary sources. Therefore, MCNPBQ used a uniform azimuthal distribution of source particle starting positions.

It should also be noted that one can (cautiously) use greater than the total amount of generated particles in an MCNP execution because the *source.f* file used to read the discrete source particle information was designed to rewind to the beginning of the SPF. Therefore, there will be five million source particle locations, directions, and energies but once the first interaction or surface is encountered, the particle will take a pseudo-unique path. If the total amount of discrete source particles used "adequately" covers the source region, directions, and energies this rewinding is valid. The limit of this adequacy is generally determined by the experience of the analyst. Unfortunately, this method cannot be extended to parallel executions of MCNP at the present time because of the serial read-rewind process.

2.4. Detector Response Function Generation

The 47-group interaction cross-section set used for the detector response functions in MCNP were generated with the Oak Ridge National Laboratory (ORNL) code GIP found in the DOORS family of codes[3]. The GIP code is part of the DOORS family that accepts nuclide-organized microscopic cross section data and prepares a group-organized file of microscopic or macroscopic cross-sections. A visual summary of both response functions is shown in Fig. 2.

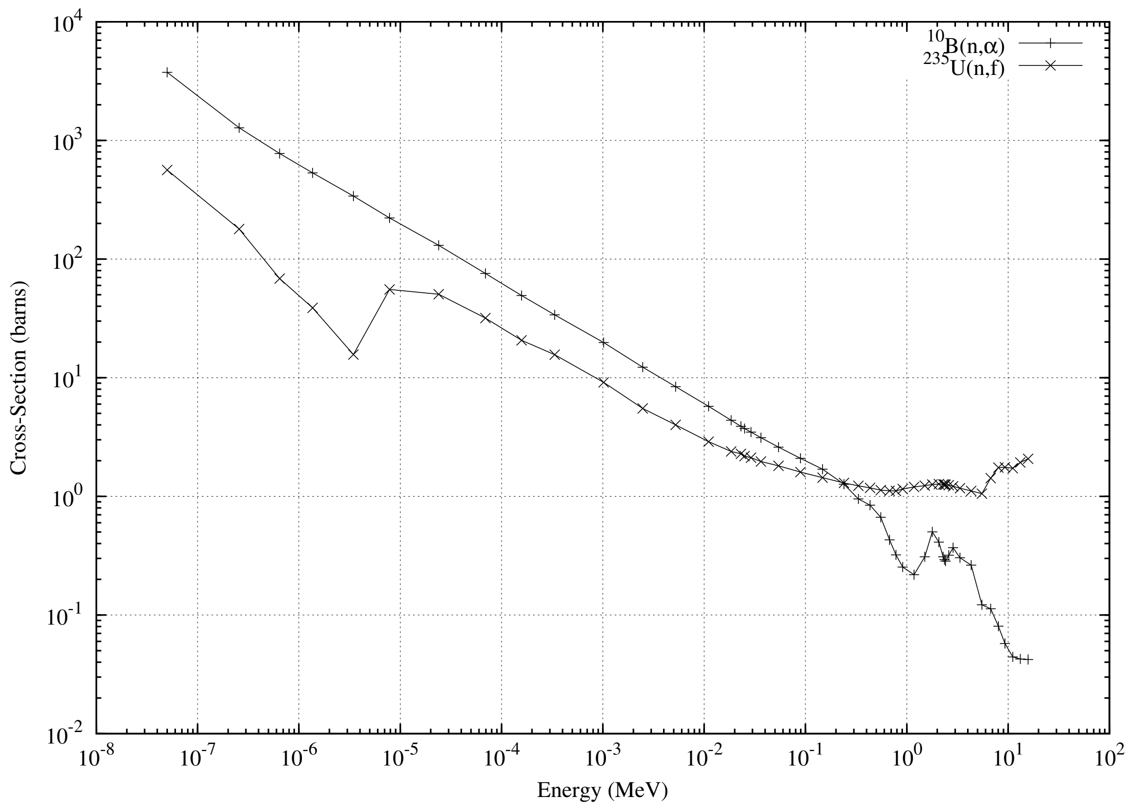

Figure 2. ^{10}B(n,α) and ^{235}U(n,f) MCNP tally response functions used to emulate BF_3 and fission chamber detector behavior.

2.5. Monte Carlo Model Development

A half reactor cavity MCNP model was created that had an azimuthal span of $[-90°, 90°]$ and an axial span from the ceiling of the room beneath the pressure vessel (where the excore detectors are loaded from) to the top of the active fuel length. Radially, the model was taken from the center of the pressure vessel out into the structural concrete to a point four feet past the excore detectors where additional depth would have only a minimal effect on the tally response. There are actually twelve excore detectors in the reactor cavity (four sets of three). However, because each detector has a negligible effect on the flux seen by neighboring detectors, only a single set was modeled. A radial schematic of the reactor cavity geometry is shown in Fig. 3. Due to the nature of the problem, extensive variance reduction wasn't required to produce statistically valid and reasonable results.

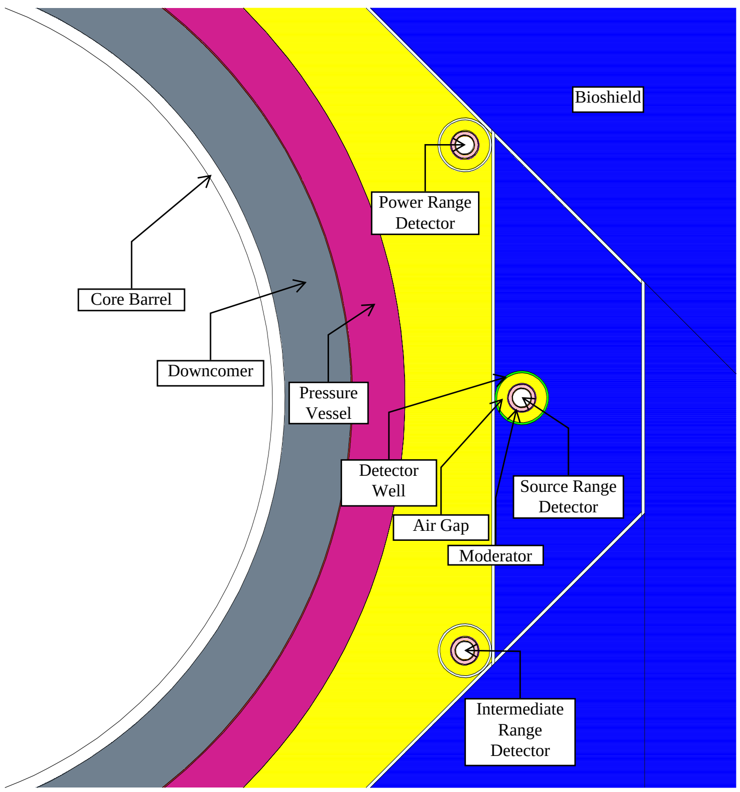

Figure 3. Radial view of the AP1000 reactor cavity geometry as modeled in MCNP.

2.5.1. Detector Geometry

Based on dimensions provided by the detector manufacturer, a standard radial distribution of material was assumed for the source, intermediate, and power range detectors which is shown in Fig. 4. It should be noted that only x_4 varied between the perturbed cases. The polyethylene moderator thicknesses used were: 0.0, 0.25, 0.50, 0.75, 1.00, 1.25, and 1.38 cm (the maximum allowed according to the detector design). The inner air gap was kept constant at 0.1 cm and the outer air gap went from the outside of the moderator to the inside of the titanium wall. The active volume of each detector was kept void in order to be used for both the BF_3 and fission chamber type of detectors. This was appropriate since the relative effect of different moderators was desired rather than an absolute approximation of detector performance.

Axially, the source and intermediate range detectors are 4′ long centered at the centerline of the fuel. The power range detector is a two-piece system. Each piece is 4′ long with a 10″ gap between the upper and lower section. The center of this gap is located at the core centerline.

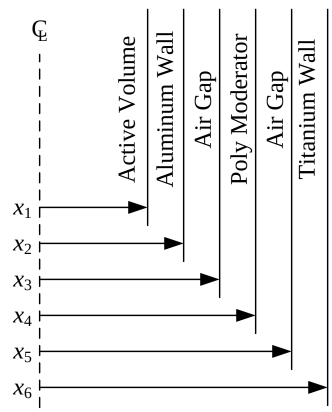

Figure 4. Cross-sectional schematic view of an excore detector.

2.5.2. Tally Summary

Each of the six tallies (two detector types for three detector locations) were normalized by a variety of parameters to make the values more physically significant. First, the flux conversion factor described in Section 2.2 was applied. Next, a volumetric multiplier was applied based on the active volume of the source, intermediate, and power range detector, as appropriate. Finally, the cross-sections sets described in Section 2.4 were applied as response functions (`de` and `df` cards). The result of these normalizations, formulated as $AV\sigma\Phi_{\text{MCNP}}$, was to produce a tally representing the volume integrated reaction rate per unit number density.

2.6. Simplistic Moderator Effect Scoping Study Summary

As mentioned previously, a simplistic model was created to quickly evaluate the effect of different moderator thicknesses and validate the bootstrapped calculation described previously. A point source of fission spectrum, mono-directional neutrons was set incident upon the center of a block of polyethylene. The moderator thicknesses analyzed were: 0.00, 0.25, 0.50, 0.75, 1.00, 1.38, 1.50, 2.50, 3.25, 4.25, 5.00, 6.50, 7.50, 8.50, 9.50, 10.00, 12.00, 14.00, 15.00, 16.00, 18.00, and 20.00 cm. All space not filled with moderator was voided.

A tally region was defined in the space next to the moderator on the side opposite the neutron source. Two tallies were taken, one each with the response functions described in Section 2.4. Only a cell-volume tally normalization was used. Therefore, the effect of normalization in this case is to give a tally signifying the volume integrated reaction rate per unit number density per MCNP source particle. As before, the value holds no significance as it is used only for relative comparisons.

A sketch of the geometry is shown in Fig. 5 where the shaded region (cell 200) is the moderator. The source is located toward the left in cell 100 and the tally region is cell 300.

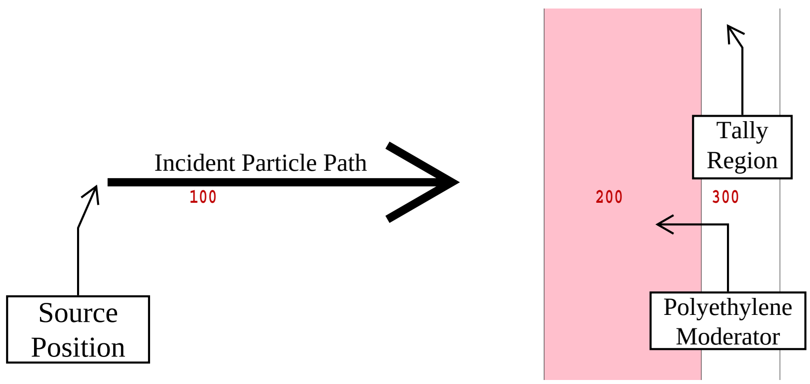

Figure 5. Simplistic scoping model featuring a fission spectrum mono-directional neutron beam incident on a block of polyethylene.

3. EXECUTION RESULTS

As indicated previously, two sets of executions were made. First, a serial execution was made on the DOMINO-MCNPBQ-MCNP model because of the limitation imposed by the MCNP discrete source particle sampling system. However, the simplistic scenario was capable of being executed on a parallel platform. The results of these executions follow.

3.1. AP1000 Excore Detector Scenario

In all cases, the reaction rates monotonically increase with moderator thickness suggesting that the moderator thickness should be maximized (at least to the investigated value of 1.38 centimeters). The results for each detector configuration for each moderator thickness are given explicitly in Tables I, II, and III. The results were also used to perform relative comparisons using a “gain ratio” (potential moderated detector performance versus an unmoderated detector) to determine the effectiveness of varying moderator thicknesses.

It should be noted that the intermediate range and power range tallies resulted in approximately equal values. While the cell volumes are different (with the power range have twice as much volume as the intermediate range) there are different total neutron populations which cause both

Table I. Tally results for both source range detector configurations.

Moderator Thickness (cm)	**BF_3**	**Gain Ratio**	**Fission Chamber**	**Gain Ratio**
0.00	$7.416\times10^{12} \pm 3.23\%$	1.00	$1.149\times10^{12} \pm 3.07\%$	1.00
0.25	$1.029\times10^{13} \pm 3.70\%$	1.39	$1.581\times10^{12} \pm 3.56\%$	1.38
0.50	$1.473\times10^{13} \pm 3.66\%$	1.99	$2.232\times10^{12} \pm 3.57\%$	1.94
0.75	$1.998\times10^{13} \pm 3.61\%$	2.69	$3.014\times10^{12} \pm 3.55\%$	2.62
1.00	$2.522\times10^{13} \pm 3.40\%$	3.40	$3.781\times10^{12} \pm 3.37\%$	3.29
1.25	$3.184\times10^{13} \pm 3.36\%$	4.29	$4.759\times10^{12} \pm 3.35\%$	4.14
1.38	$3.561\times10^{13} \pm 3.27\%$	4.80	$5.323\times10^{12} \pm 3.26\%$	4.63

Table II. Tally results for both intermediate range detector configurations.

Moderator Thickness (cm)	**BF_3**	**Gain Ratio**	**Fission Chamber**	**Gain Ratio**
0.00	$9.034\times10^{12} \pm 2.38\%$	1.00	$1.496\times10^{12} \pm 2.12\%$	1.00
0.25	$1.556\times10^{13} \pm 2.89\%$	1.72	$2.445\times10^{12} \pm 2.70\%$	1.63
0.50	$2.614\times10^{13} \pm 2.86\%$	2.89	$3.990\times10^{12} \pm 2.77\%$	2.67
0.75	$4.145\times10^{13} \pm 2.60\%$	4.59	$6.224\times10^{12} \pm 2.57\%$	4.16
1.00	$5.607\times10^{13} \pm 2.49\%$	6.21	$8.388\times10^{12} \pm 2.47\%$	5.61
1.25	$7.020\times10^{13} \pm 2.36\%$	7.77	$1.048\times10^{13} \pm 2.35\%$	7.01
1.38	$7.861\times10^{13} \pm 2.29\%$	8.70	$1.173\times10^{13} \pm 2.29\%$	7.84

Table III. Tally results for both power range detector configurations.

Moderator Thickness (cm)	**BF_3**	**Gain Ratio**	**Fission Chamber**	**Gain Ratio**
0.00	$9.145\times10^{12} \pm 1.81\%$	1.00	$1.514\times10^{12} \pm 1.62\%$	1.00
0.25	$1.629\times10^{13} \pm 2.20\%$	1.78	$2.540\times10^{12} \pm 2.07\%$	1.68
0.50	$2.722\times10^{13} \pm 2.11\%$	2.98	$4.126\times10^{12} \pm 2.05\%$	2.73
0.75	$3.993\times10^{13} \pm 1.93\%$	4.37	$6.003\times10^{12} \pm 1.91\%$	3.96
1.00	$5.728\times10^{13} \pm 1.80\%$	6.26	$8.561\times10^{12} \pm 1.79\%$	5.65
1.25	$7.087\times10^{13} \pm 1.68\%$	7.75	$1.057\times10^{13} \pm 1.68\%$	6.98
1.38	$7.864\times10^{13} \pm 1.65\%$	8.60	$1.173\times10^{13} \pm 1.64\%$	7.75

tallies to result in an approximately equal average, track-length calculated, flux value. Furthermore, the detectors are in symmetric locations relative to the pressure vessel so it is expected that they will be in virtually identical neutron environments. As such, both detectors showing a near identical response is reasonable and expected.

3.2. Simplistic Moderator Effect Scoping Study Scenario

With the input and methods described in Section 2.6, 22, one million source neutron parallel MCNP executions (one for each moderator thickness considered) were made.

A graphical representation of the results (with appropriate uncertainty bars) is given in Fig. 6. Similar to before, the results were used to perform relative comparisons using a “gain ratio” (potential moderated detector performance versus an unmoderated detector) to determine the effectiveness of varying moderator thicknesses. As expected the reaction rate reached a peak, in this case, at 7.5 cm (gain ratio of 1381 and 79 for BF_3 and fission chamber, respectively). Thus, in this case the optimum moderator thickness would be 7.5 centimeters with additional moderator acting as a shield. Therefore, this value is on the order of where one might encounter the overall optimum thickness for the AP1000 excore detector moderator without any mechanical constraints due to a detector housing.

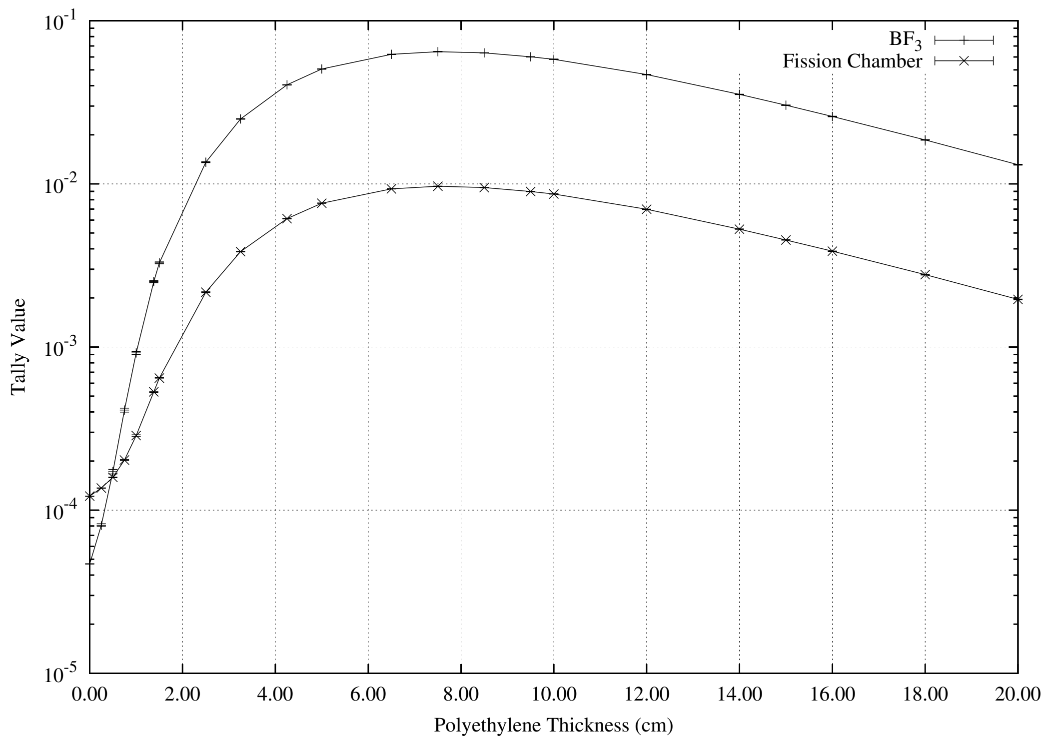

Figure 6. Cross-sectional schematic view of an excore detector.

4. CONCLUSIONS

Based on the coupled deterministic-stochastic analysis of the geometry immediately surrounding the AP1000 pressure vessel and excore detector system, it was recommended that polyethylene moderator be included to the maximum allowable extent, approximately 1.38 cm based on detector design specifications, to improve signal strength due to increased reaction rates in the active detector volume. Moreover, if metallic shielding is added (to protect the detector from post-shutdown gamma radiation) the moderator will become more effective because of the reduction in the thermal neutron population by the shield.

If the detector's mechanical design changes significantly, the effect of a thicker moderator should be reanalyzed. However, based on the simplistic scoping study performed it was expected that at a thickness of roughly 7.5 cm, a peak reaction rate will be achieved and at greater thicknesses the moderator will act as a shield reducing effective signal strength. Therefore, increasing the moderator thickness until approximately 7.5 cm should provide additional gains in signal strength for either the BF_3 or fission chamber detectors.

As far as the methodology used in performing this calculation, it seems clear that the coupling of deterministic-stochastic transport methods allows for more detailed and realistic sources to be developed and used in production calculations. To accurately represent the source flux spectrum being emitted from the pressure vessel outside surface to the surrounding environment, deterministic transport methods are the only option for a calculation of reasonable duration. However, without significant mesh refinement, it would be impossible to capture the effects of different moderator sizes on the order of centimeters. Unfortunately, there are still some limitations to the method used. Chiefly, DOMINO only operates with r-z DORT geometry so cylindrical and disc sources are azimuthally uniform. Therefore, the core loading pattern was homogenized and the intermediate and power range detectors saw virtually identical environments. Furthermore, when considering other types of analyses, this method cannot be used for generating linear sources, such as one might find in coolant within a pipe. Efforts are currently being planned to extend this method to full 3D capability.

REFERENCES

[1] S. M. Girard (ed.), *MCNP — A General Monte Carlo N-Particle Transport Code, Version 5*, RSICC, Oak Ridge, TN, USA (2003).

[2] R. Orsi, *BOT3P Version 5.2: A Pre/Post-Processor System for Transport Analysis*, ENEA, Centro Richerche Bologna, Italy (2007).

[3] W. A. Rhoades, et al., *DOORS3.2 — One, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System*, RSICC, Oak Ridge, TN, USA (1998).

[4] R. W. Roussin, et al., *BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications*, RSICC, Oak Ridge, TN, USA (1996).