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Shielding Analysis and Design of the KIPT Experimental Neutron Source Facility of Ukraine

Nuclear Engineering Division

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Abstract

Argonne National Laboratory (ANL) of USA and Kharkov Institute of Physics and Technology (KIPT) of Ukraine have been collaborating on the conceptual design development of an experimental neutron source facility based on the use of an electron accelerator driven subcritical (ADS) facility [1]. The facility uses the existing electron accelerators of KIPT in Ukraine. The neutron source of the sub-critical assembly is generated from the interaction of 100 KW electron beam with a natural uranium target. The electron beam has a uniform spatial distribution and the electron energy in the range of 100 to 200 MeV, [2]. The main functions of the facility are the production of medical isotopes and the support of the Ukraine nuclear power industry. Reactor physics experiments and material performance characterization will also be carried out.

The subcritical assembly is driven by neutrons generated by the electron beam interactions with the target material. A fraction of these neutrons has an energy above 50 MeV generated through the photo nuclear interactions. This neutron fraction is very small and it has an insignificant contribution to the subcritical assembly performance. However, these high energy neutrons are difficult to shield and they can be slowed down only through the inelastic scattering with heavy isotopes. Therefore the shielding design of this facility is more challenging relative to fission reactors. To attenuate these high energy neutrons, heavy metals (tungsten, iron, etc.) should be used. To reduce the construction cost, heavy concrete with 4.8 g/cm^3 density is selected as a shielding material. The iron weight fraction in this concrete is about 0.6. The shape and thickness of the heavy concrete shield are defined to reduce the biological dose equivalent outside the shield to an acceptable level during operation. At the same time, special attention was give to reduce the total shield mass to reduce the construction cost. The shield design is configured to maintain the biological dose equivalent during operation ≤ 0.5 mrem/h inside the subcritical hall, which is five times less than the allowable dose for working forty hours per week for 50 weeks per year.

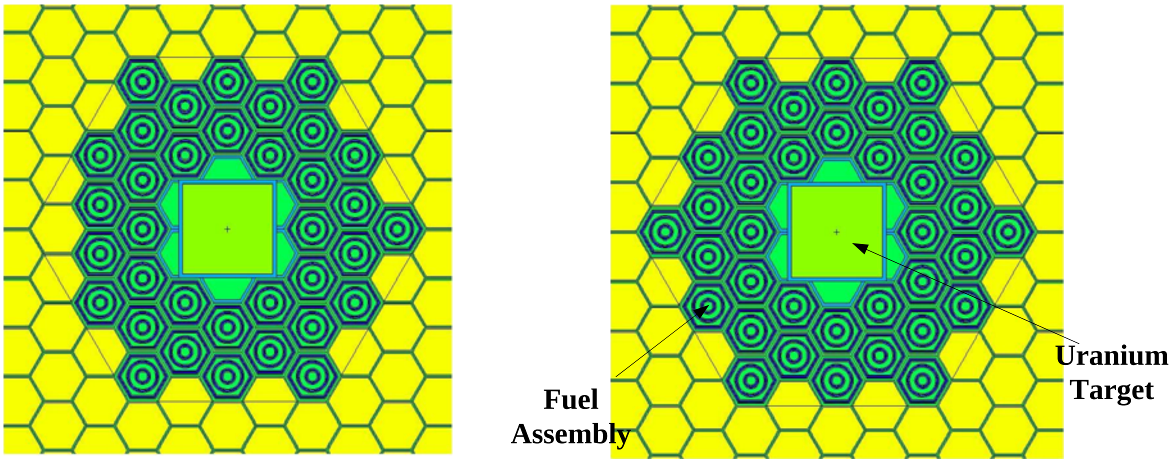
This study analyzed and designed the thickness and the shape of the radial and top shields of the neutron source based on the biological dose equivalent requirements inside the subcritical hall during operation. The Monte Carlo code MCNPX is selected because of its capabilities for transporting electrons, photons, and neutrons. Mesh based weight windows variance reduction technique is utilized to estimate the biological dose outside the shield with good statistics. A significant effort dedicated to the accurate prediction of the biological dose equivalent outside the shield boundary as a function of the shield thickness without geometrical approximations or material homogenization. The building wall was designed with ordinary concrete to reduce the biological dose equivalent to the public with a safety factor in the range of 5 to 20.

I. Introduction

Both national and international research institutions are considering accelerator driven systems (ADS) in their fuel cycle scenarios for transmuting actinides and long-lived fission products. Therefore, several studies and experiments have been performed using accelerator driven subcritical systems. As a part of the collaboration activity between the United States of America and Ukraine, Argonne National Laboratory (ANL) and the National Science Center-Kharkov Institute of Physics and Technology (NSC-KIPT) have been collaborating on developing a neutron source facility based on the use of electron accelerator driven subcritical system. The main functions of this facility are the medical isotope production and the support of the Ukraine nuclear industry. Physics experiments and material research will also be carried out utilizing the sub-critical assembly. KIPT did have a plan to construct this facility using high-enriched uranium (HEU) fuel. These collaborative studies showed that the use of low enriched uranium (LEU) instead of HEU enhances the facility performance and the main system choices and design parameters were defined [2].

The developmental analyses defined the geometry of the subcritical assembly, the target assembly design, and its location for producing the neutron source, the fuel loading, the reflector material and thickness, and the facility performance parameters [2]. The fuel design is WWR-M2 type, which is used for Kiev research reactor [3] and other test reactors with water coolant. It has a hexagonal geometry with 3.5 cm pitch. The fuel material is uranium oxide in an aluminum matrix and aluminum clad with 50 cm active height. The U-235 enrichment is ≤ 20%. The subcritical assembly has 35 to 36 fuel assemblies surrounded by graphite reflector inside a water tank. The electron interactions with the target material produce high energy photons, which generate neutrons through photonuclear reactions with the target material. Such interactions occur at the center of the subcritical assembly and the produced neutrons drive the subcritical assembly. The radial configuration of the subcritical assembly is shown in Figure 1, which includes the target, the fuel assemblies, and the reflector assemblies.

The shielding study was carried out to define the biological dose equivalent outside the biological shield of the facility in the radial and vertical directions during operation as a function of the shield geometry and thickness. The main objective is to define a shielding configuration, which minimizes the irradiation exposure as much as possible and permits the personnel to work around the subcritical assembly during operation. The shield design is configured to maintain the biological dose during operation ≤ 0.5 mrem/h, which is five times less than the allowable dose to work forty hours per week for 50 weeks per year. In the shielding design process, cost and engineering considerations were also considered. The wall thickness of the facility building was defined based on the international limit for the biological dose equivalent with a safety factor in the range of 5 to 20.



35 fuel assemblies **36 fuel assemblies**

Figure 1. Radial configuration of the subcritical assembly

The shielding study requires accurate characterization of the neutrons flux through the shield. This requires electrons, neutrons, and photons transport through the main components including the target, the fuel, the reflector, and the shield. The electron interactions with the target material produce high-energy photons, which generate neutrons through photonuclear reactions with the target material. Such process occurs at the center of the assembly and the produced neutrons reach the nuclear fuel for driving the sub-critical assembly.

The Monte Carlo computer code MCNPX [4] was used with ENDF/B-VI nuclear data libraries for performing the analysis. To achieve the required biological dose equivalent level, the shield configuration has to have a neutron attenuation factor greater than 10^{13}. In addition for the electron driven facility, the neutron yield is only about 0.1 neutrons per electron [5]. This represents a major challenge for the analogue Monte Carlo calculation to get an accurate estimate of the neutron flux outside the biological shield since the number of neutron tracks outside is less than one for every 10^{14} sampled electrons. The computing time for such problem using analogue Monte Carlo is impractical. The weight windows variance reduction technique was utilized to perform this shielding task, in addition to the developed methodology of this study.

II. Shielding Analysis of the Radial Shield Section of the Subcritical Assembly

The radial section of the biological shield has a cylindrical geometry surrounding the subcritical assembly. Only, steel and heavy concrete (referred to as concrete) were selected as shielding materials to reduce the cost. Three dimensional models have been developed for performing the radial shielding analyses. MCNPX, the Monte Carlo transport computer code, is used to transport the transport of electrons and the generated photons and neutrons. The target configuration, fuel assemblies, and graphite zones are explicitly modeled without any geometrical approximation to get an accurate prediction for the neutron flux.

The international guideline for the biological dose equivalent in controlled working area is 2.5-mrem/hr for 40-hours working week. In this analysis, a safety factor of 5.0 is adopted to account for uncertainties in the nuclear data, imperfections in the geometrical model, and calculation method approximations. Therefore, a biological dose limit of 0.5-mrem/hr is used to define the required shield configuration in this study.

II.1 Simplified Geometrical Radial Shield Model

The first set of analyses was performed to define the required shield thickness to satisfy the dose limit based on a simplified geometrical model. In addition, the study examined the performance of a shield design consisting of steel followed by concrete. The simplified model has a cylindrical representation for the electron beam, the target, and the beryllium reflector, the water tank, and the biological shield. The outer radius of the beryllium reflector is 60 cm. The fuel section is presented explicitly without any geometrical approximation or material homogenization. The beryllium reflector is homogenized to reduce the computational time. Figure 2 shows the geometrical model. In this model, the details of the top section of the facility are not included since it does not affect the performance of the radial shield.

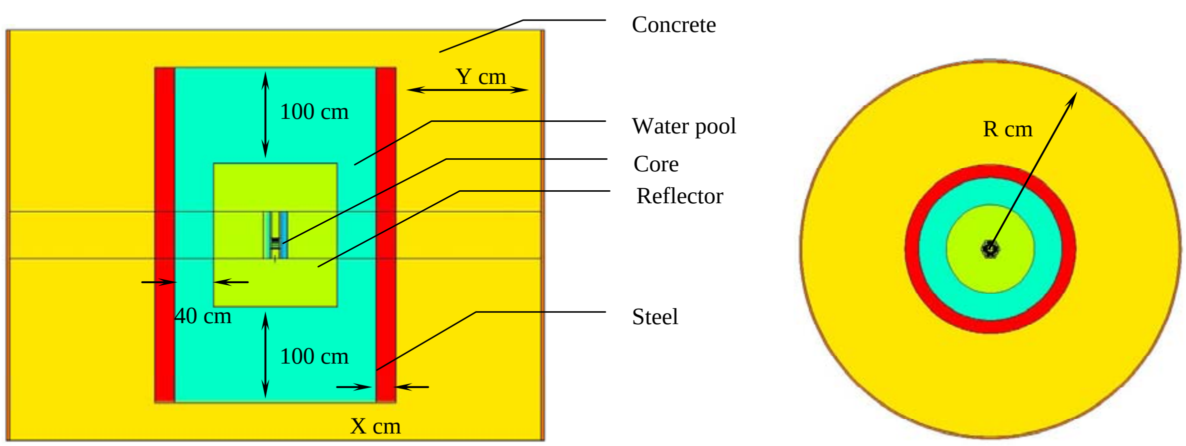


Figure 2. Simplified geometrical model for the radial shielding analysis

II.2 Results & Analysis from the Simplified Geometrical Radial Shield Model

The goal of this study is to define the radial shield thickness that reduces the biological dose equivalent to 0.50-mrem/hr for different combination of steel and concrete materials. The MCNPX code with appropriate variance reduction techniques was used to estimate the flux level outside of the reactor shield boundary. This flux is folded with ICPR-21 (1971) flux-to-dose conversion tables to obtain the corresponding biological dose equivalent form neutrons and photons. The concrete (heavy concrete) has a density of 4.8 g/cm^3. The steel was used in front of the concrete and its thickness was varied in steps in the range 0 to 20 to study the impact on the shielding performance. A sample of the results is shown in Table I for 200 MeV electron beam. At the outer surface of the radial shield, the results show that the photon biological dose equivalent is about four orders of magnitude less than the neutron biological dose equivalent. Therefore, the shielding analyses for such facilities have to give more attention to the neutron biological dose. The use of 170-cm concrete shield, 276-cm outer radius, results in a total dose of 0.9 mrem/h. For a fixed total shield thickness, the use of a steel layer with the heavy concrete shield did not enhance the shield performance but it increases its cost. For a total shield thickness of 170-cm, the use 20 cm of steel reduces the total biological dose equivalent from 0.9 to 0.67 mrem/h. Therefore, the concrete shield design is used to satisfy the limit for the biological dose equivalent.

Table I Neutron and photon biological doses as a function of the shield thickness

Shield thickness		Shield Radius cm	Neutron Dose (mrem/hr)	σ, %	Photon Dose (mrem/hr)	σ, %
Steel, X cm	Concrete, Y cm					
0	137.5	243	5.45	9.2	7.15E-04	8.2
0	170	276	0.90	9.8	1.14E-04	8.9
10	140	256	2.13	8.8	2.73E-04	7.8
10	150	266	1.36	12.3	1.70E-04	10.9
10	160	276	0.84	12.9	1.04E-04	11.5
20	130	256	1.81	11.2	2.27E-04	10.0
20	140	266	1.05	12.1	1.31E-04	10.5
20	150	276	0.67	12.3	8.27E-05	11.0

For the 100 KW electron beam power, the results of Table II and Table III show the biological dose equivalent outside the shield surface for different electron energies. As the electron energy increases, the energy of the generated photons increases. Consequently, the fraction of the very high energy neutrons (> 20 MeV) generated through the photonuclear reactions increases and the biological dose equivalent increases. In addition, the biological dose equivalent produced by neutrons is much larger than that produced by photons.

For the subcritical assembly design with a graphite reflector and 170-cm concrete shield, the outside neutron dose is 0.8 mrem/hr for neutrons and 1e-4 mrem/hr for photons. These biological dose values are very close to the corresponding values obtained with the beryllium reflector for the same concrete shield thickness. Therefore, using different reflector material has little effect on the shielding performance. In fact, the very high energy neutrons from the photonuclear reactions, which represent a very small fraction of the total neutron yield, are responsible for the biological dose equivalent outside the shield.

Table II. Neutron fluxes at different locations for 100 KW electron beam with electron energy of 100, 150, or 200 MeV

Upper energy limit (MeV)	Outer reflector surface		
	100 MeV	150 MeV	200 MeV
1.00e-03	1.09e+08	1.06e+08	1.00e+08
1.00e-02	1.20e+06	1.13e+06	1.37e+06
1.00e-01	1.40e+06	1.38e+06	1.30e+06
5.00e-01	1.41e+06	1.51e+06	1.20e+06
1.00e+00	7.61e+05	7.33e+05	6.95e+05
2.00e+00	8.83e+05	9.99e+05	8.79e+05
5.00e+00	8.93e+05	7.65e+05	8.75e+05
1.00e+01	3.93e+05	3.63e+05	4.00e+05
2.00e+01	2.02e+05	2.21e+05	2.42e+05
2.00e+02	9.10e+04	2.11e+05	3.47e+05
Total	1.16e+08	1.13e+08	1.07e+08
Upper energy limit (MeV)	Inner concrete surface		
	100 MeV	150 MeV	200 MeV
1.00e-03	7.67e+05	6.94e+05	1.06e+06
1.00e-02	1.29e+05	1.17e+05	2.02e+05
1.00e-01	1.91e+05	1.74e+05	1.69e+05
5.00e-01	3.27e+05	2.93e+05	3.56e+05
1.00e+00	2.71e+05	2.28e+05	2.36e+05
2.00e+00	2.49e+05	2.17e+05	2.15e+05
5.00e+00	1.91e+05	1.72e+05	1.67e+05
1.00e+01	4.12e+04	3.78e+04	4.06e+04
2.00e+01	8.37e+03	1.08e+04	1.29e+04
2.00e+02	6.83e+03	2.01e+04	3.63e+04
Total	2.18e+06	1.96e+06	2.49e+06
Upper energy limit (MeV)	Outer concrete surface		
	100 MeV	150 MeV	200 MeV
1.00e-03	7.30e-02	8.22e-01	2.47e+00
1.00e-02	1.91e-02	2.24e-01	6.69e-01
1.00e-01	2.85e-02	3.40e-01	1.02e+00
5.00e-01	4.41e-02	5.32e-01	1.61e+00
1.00e+00	3.91e-02	4.79e-01	1.46e+00
2.00e+00	3.79e-02	4.75e-01	1.46e+00
5.00e+00	3.69e-02	4.92e-01	1.54e+00
1.00e+01	1.87e-02	2.40e-01	7.48e-01
2.00e+01	1.98e-02	2.21e-01	6.58e-01
2.00e+02	1.07e-01	1.60e+00	5.10e+00
Total	4.21e-01	5.43e+00	1.67e+01

* The maximum statistical error is about 10%, which happens at the outer surface of shield

The generated neutron flux was also characterized as a function of the electron beam energy for the 100 KW beam power. The shield configuration with 10 cm steel and 150 cm concrete is used for this analysis. The energy-dependent neutron fluxes were calculated at different locations, the outer surface of reflector, and the inner and outer surface of radial shield respectively. The results are plotted in Figure 3 through 5. At the reflector boundary, the neutron flux values produced with different electron energies are almost the same except for the neutrons with energy above 20 MeV. These neutrons penetrate the concrete shield and they are responsible for the biological dose outside the shield as can be seen from Figure 5. The neutron spectra of Figure 6 show clearly that the high electron energy requires extra shielding. The high energy neutrons penetrate the shield and increase the flux level as shown in Figure 5 although they represent a small fraction of the total neutron yield. Therefore, it can be concluded that the biological dose equivalent outside the shield is dominantly caused by the high energy neutrons (> 20 MeV) generated in the target and the shield performance for slowing down these very high energy neutrons. The fission neutrons of the subcritical assembly (< 10 MeV) have very small contribution to the biological dose outside the shield.

The results in Table III show that the biological dose equivalent outside the radial shield from the 200 MeV electrons is ~50 times larger than that from the 100 MeV electrons. Although the electron beam power is fixed at 100 kW for the two sets of the electrons. This result is consistent with the results in Table II and Figure 6.

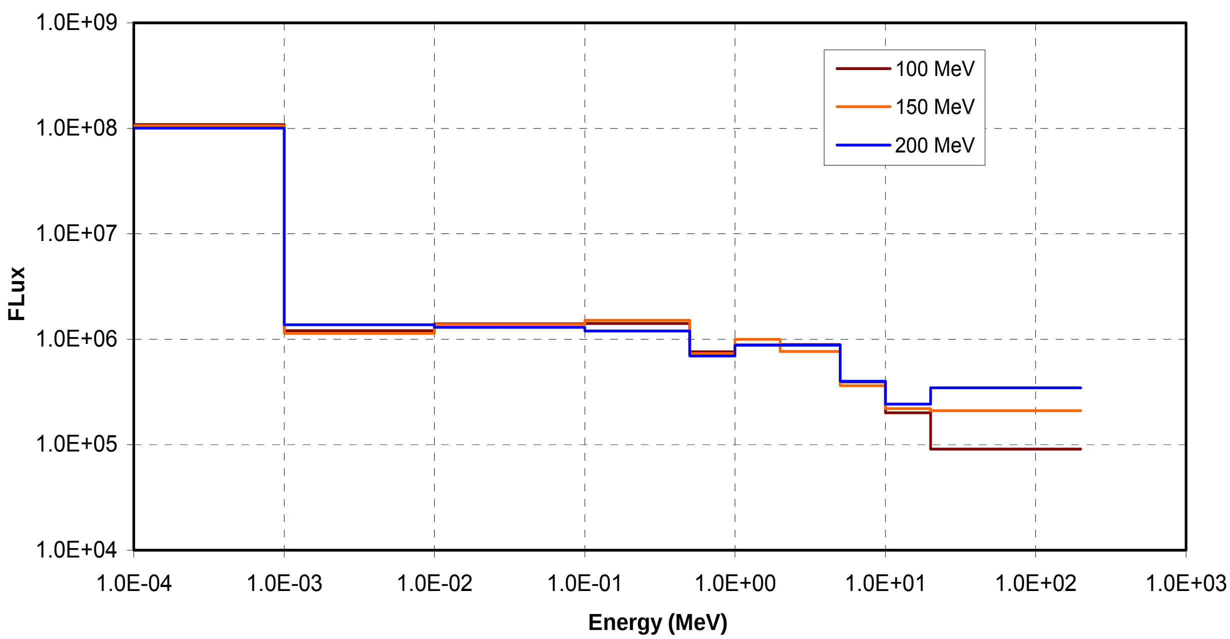


Figure 3. Neutron fluxes at the outer surface of reflector for 100 KW electron beam with electron energy of 100, 150, or 200 MeV

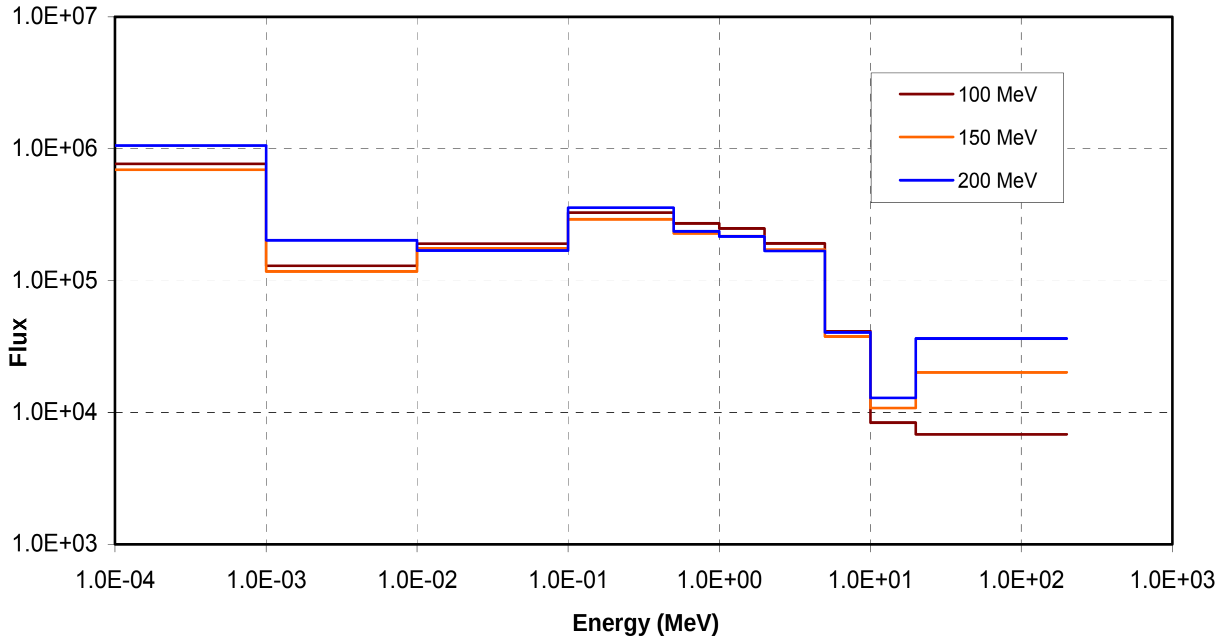


Figure 4. Neutron fluxes at the inner surface of the concrete shield for 100 KW electron beam with electron energy of 100, 150, or 200 MeV

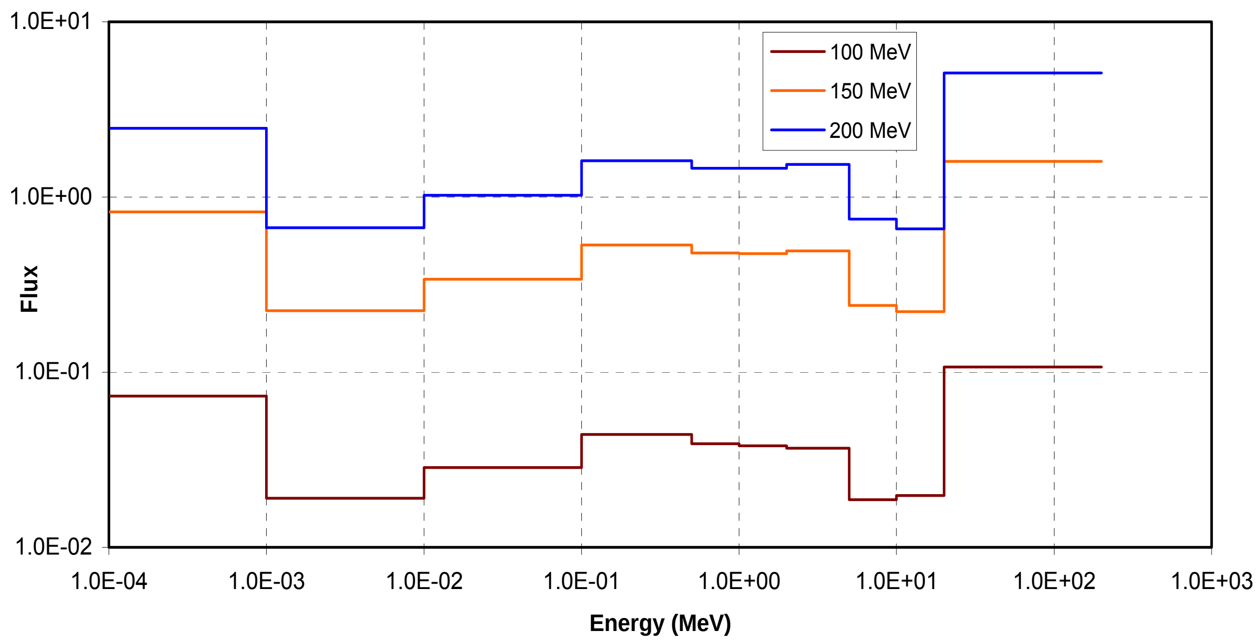


Figure 5. Neutron fluxes at the outer surface of the concrete shield for 100 KW electron beam with electron energy of 100, 150, or 200 MeV

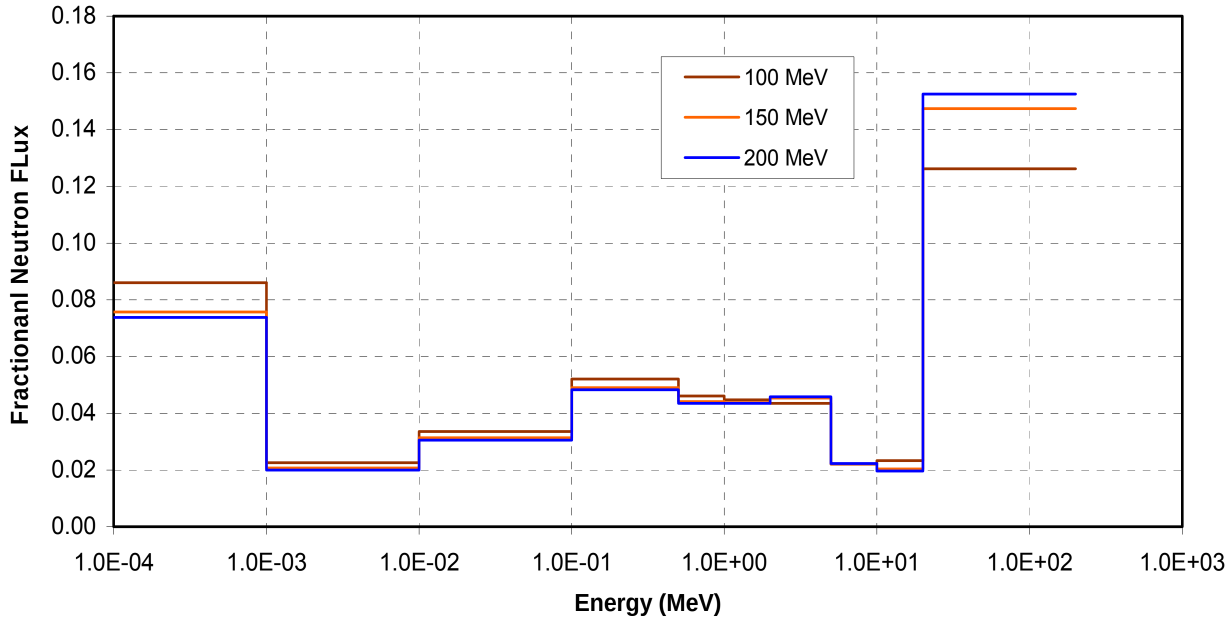


Figure 6. Neutron spectra at the outer surface of the concrete shield for 100 KW electron beam with electron energy of 100, 150, or 200 MeV

Table III. Comparison of neutron dose outside the radial shield for the cases using 100 MeV, 150 MeV and 200 MeV electron beams

Upper energy limit (MeV)	100 MeV		150 MeV		200 MeV	
	Neutron dose (mrem/hr)	σ, %	Neutron dose (mrem/hr)	σ, %	Neutron dose (mrem/hr)	σ, %
1.00e-03	1.90e-04	8.2	2.13e-03	9.1	6.38e-03	5.4
1.00e-02	4.27e-05	8.4	5.02e-04	9.3	1.50e-03	15.3
1.00e-01	2.27e-04	8.4	2.70e-03	9.3	8.11e-03	3.3
5.00e-01	1.32e-03	8.5	1.59e-02	9.4	4.82e-02	2.9
1.00e+00	2.28e-03	8.7	2.79e-02	9.4	8.45e-02	4.4
2.00e+00	3.07e-03	8.8	3.81e-02	9.5	1.16e-01	3.5
5.00e+00	3.32e-03	9.0	4.39e-02	9.8	1.36e-01	9.2
1.00e+01	1.77e-03	8.9	2.24e-02	9.7	6.91e-02	3.5
2.00e+01	2.00e-03	8.1	2.20e-02	8.9	6.47e-02	1.9
2.00e+02	1.37e-02	10.7	2.15e-01	10.6	6.90e-01	1.9
Total	2.79e-02	9.64	3.90e-01	10.1	1.23e+00	5.1

II.3 Detailed Geometrical Radial Shield Model

Based on the previous results, a detailed geometrical model was developed using the latest design features. The shielding material is only concrete, heavy concrete, with 4.8 g/cm^3 density, with a thickness in the range of 150~180 cm. The subcritical assembly has a graphite reflector and its height is 60 cm. The radius of water tank is 100 cm as shown in Figure 7. The subcritical assembly has 36 fuel assemblies, with a square target located at the center. The top cover is neglected because it does not affect the radial shielding analyses. The electron beam has a uniform distribution and a square cross section. The geometrical structural details of all the components are included explicitly in the calculational model.

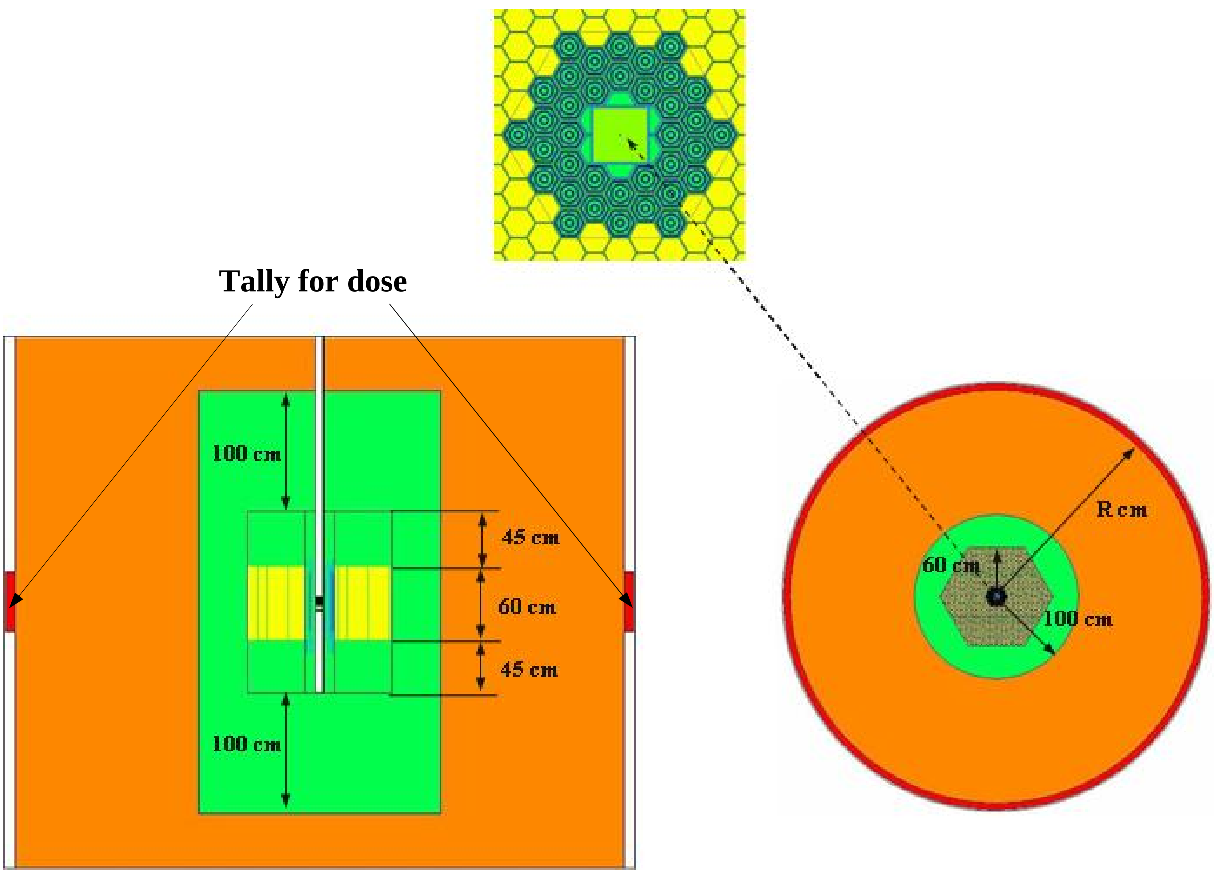


Figure 7. MCNPX geometrical model for the radial shielding analysis

II.4 Results & Analysis from the Detailed Geometrical Radial Shield Model

As discussed before, direct analogue Monte Carlo calculation is impractical for this shielding problem. Variance reduction technique must be used to perform the analyses. The mesh based weight windows generation capability of MCNPX was utilized for the calculations. The weight windows are based on space-energy-dependent splitting and Russian roulette techniques. For each space-energy phase cell, a lower and upper weight bounds are specified for the tracked particle or photon. These weight bounds define a window of acceptable weights for the tracked particle or photon. If the weight of the tracked particle or photon is below the lower weight bound, Russian roulette is carried out, and the weight of tracked particle or photon is either adjusted to a value within the weight window or the tracked particle or photon is terminated. If the weight of the particle or photon is above the upper weight bound, it is split so that the individual weight of each split particle or photon is within the weight window. No action is taken for particles or photons with weight within the weight windows [6].

MCNPX computer code has a weight windows generator which can generate these importance functions. The task of choosing importance is simplified and insight information about the shielding problem can be obtained with this capability. The geometrical model is divided into a number of phase space cells, the importance of a cell then can be defined as the score generated by a unit weight particle after entering the cell. Thus, the cell's importance can be estimated as the total score from particles entering the cell divided by the total weight entering the cell. MCNPX assigns weight windows inversely proportional to the importance.

In this study, a weight windows mesh is utilized to eliminate the need to subdivide the geometrical model into a large number of regions. A neutron dose tally is used outside the radial shield as shown in Figure 7, and the weight windows were generated for this tally. Sometimes, the weight windows generator produces bad estimates of the importance function because of the statistical nature of the generator. In particular, unless a phase space region is sampled adequately, there will be either no importance estimate, i.e. zero values or unreliable values [6]. For the mesh based weight windows generator, a very coarse mesh will produce a crude estimate of the importance function, while a very fine mesh will produce zero values due to insufficient tracks per cell.

Due to the symmetry of the geometry for the radial shield, R-Z mesh is utilized. The spatial meshes are azimuthally homogenous, thus it's relatively easier to get sufficient samples. The weight windows generated for the simplified geometry, as discussed above, could be used as a good initial guess, thus the weight windows for the new geometry could be converged faster.

The energy-dependent neutron weight windows for the final calculations are plotted in Figure 8 through 10. From the plots of the weight windows, first, it can be seen that the low energy neutrons (< 10 MeV) generated directly in the core region make very little contributions to the dose outside the shield. For the high energy neutrons (> 10 MeV) generated in the target region, although their fraction is very small, (for the 200 MeV electron beam case, its fraction is less than 0.01%, for the 100 MeV electron beam case, its fraction is even much smaller), their contributions to the dose outside the shield is dominant.

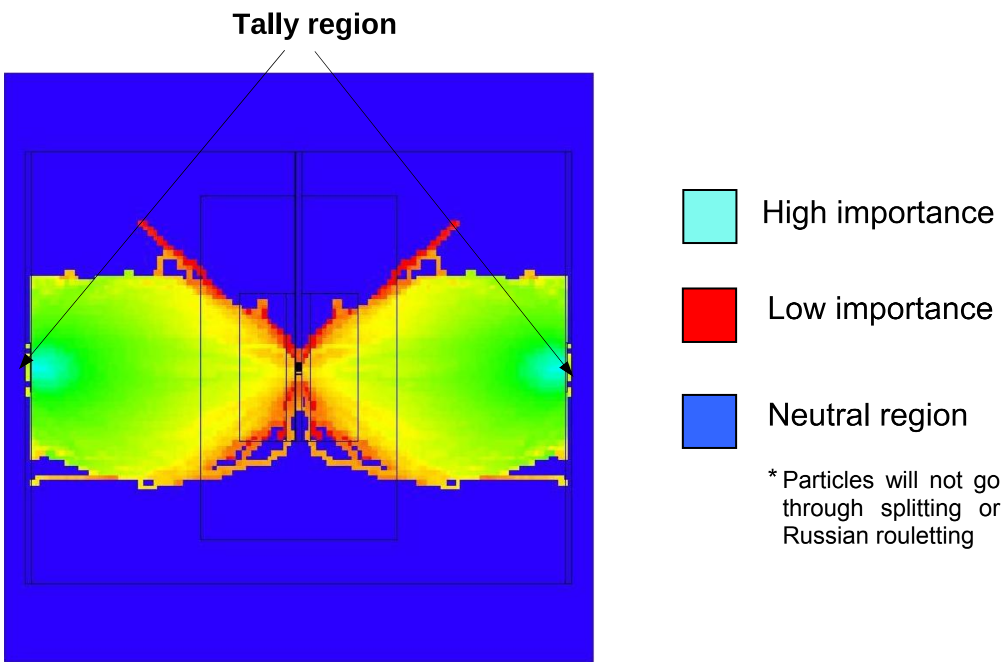


Figure 8. Weight Windows for the neutrons with energy > 10 MeV

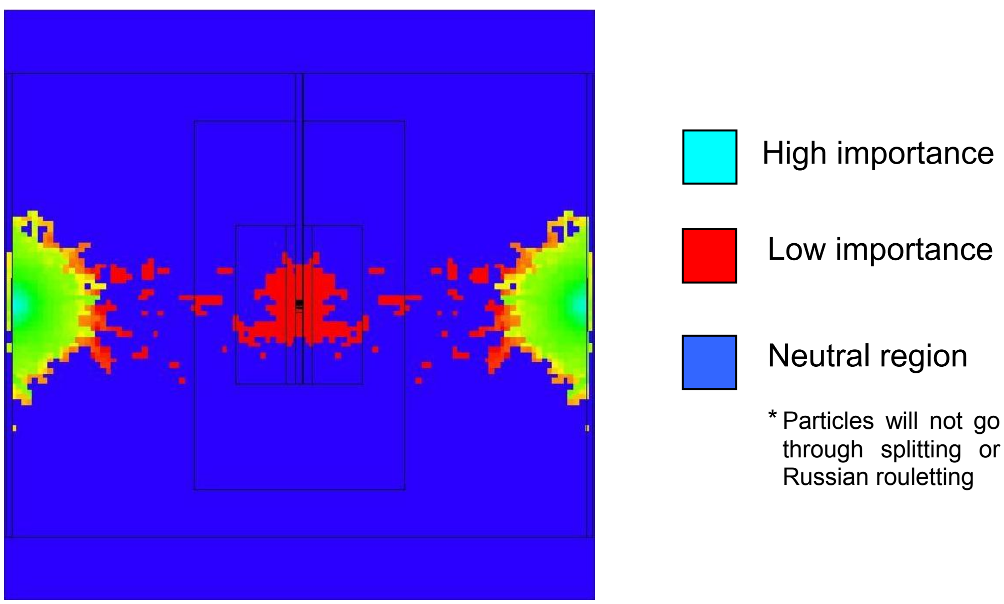


Figure 9. Weight Windows for the neutrons with energy between 0.1 MeV and 10 MeV

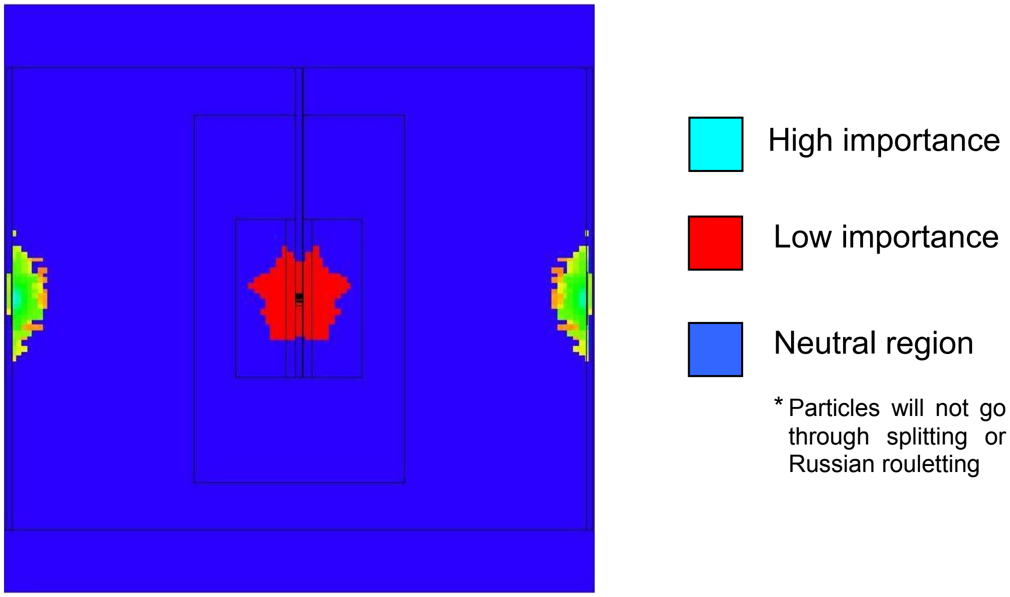


Figure 10. Weight Windows for the neutrons with energy < 0.1 MeV

In the MCNPX results, the tallied flux is folded with ICPR-21 (1971) flux-to-dose equivalent conversion tables to obtain the corresponding neutron and photon biological dose equivalent. For the case using 200 MeV electron beam, the calculated neutron and photon dose outside the radial shield with different thickness of concrete are given in Table IV.

Table IV. Neutron and photon biological dose outside the shield surface for different concrete shield thicknesses

Concrete Thickness (cm)	Shield Radius cm	Neutron Dose (mrem/hr)	σ, %	Photo Dose (mrem/hr)	σ, %
152.0	252.0	2.43	12.6	4.59E-04	12.2
162.0	262.0	1.39	12.6	2.50E-04	12.0
172.0	272.0	0.72	11.8	1.58E-04	17.3
182.0	282.0	0.48	11.1	8.32E-05	10.7

The neutron dose maps are also calculated for different shield thicknesses of concrete, using the mesh tally capability of MCNPX. The results are shown in Figure 11 through 14.

Based on the results shown in Table IV and Figure 11 through 14, the use of 182-cm concrete shield thickness reduces the biological dose equivalent outside the shield surface to less than 0.5 mrem/hr.

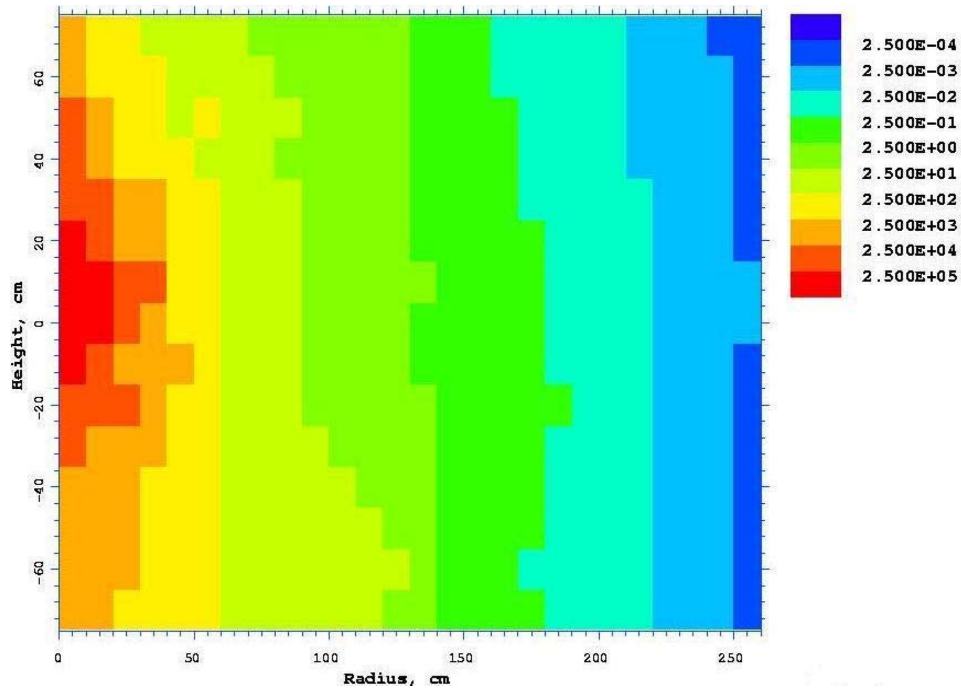


Figure 11. Neutron Dose map resulted from the use of 152-cm concrete shield, rem/hr

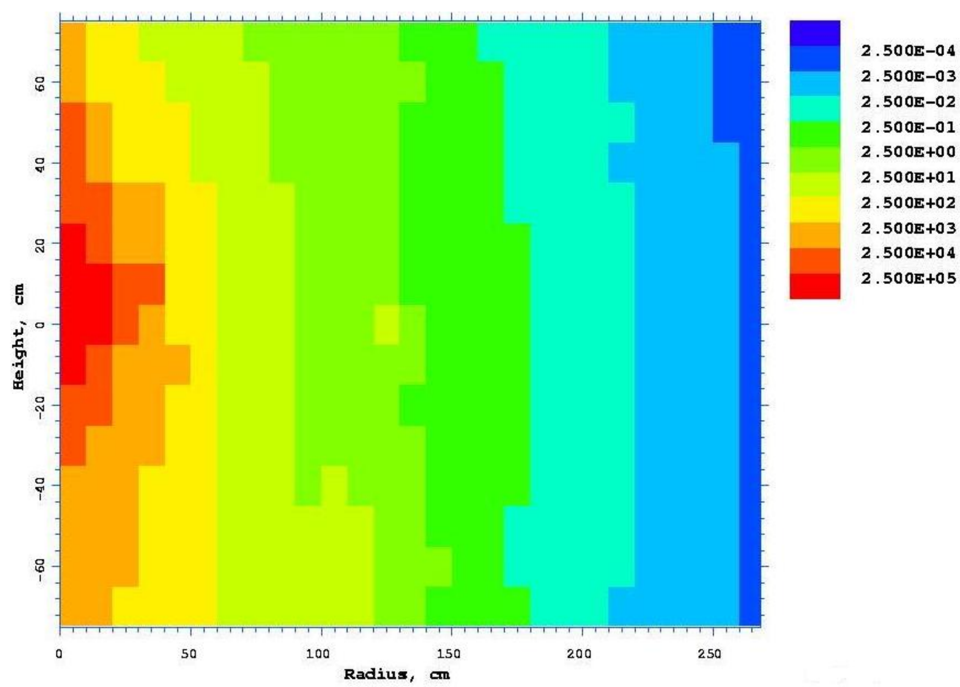


Figure 12. Neutron Dose map resulted from the use of 162-cm concrete shield, rem/hr

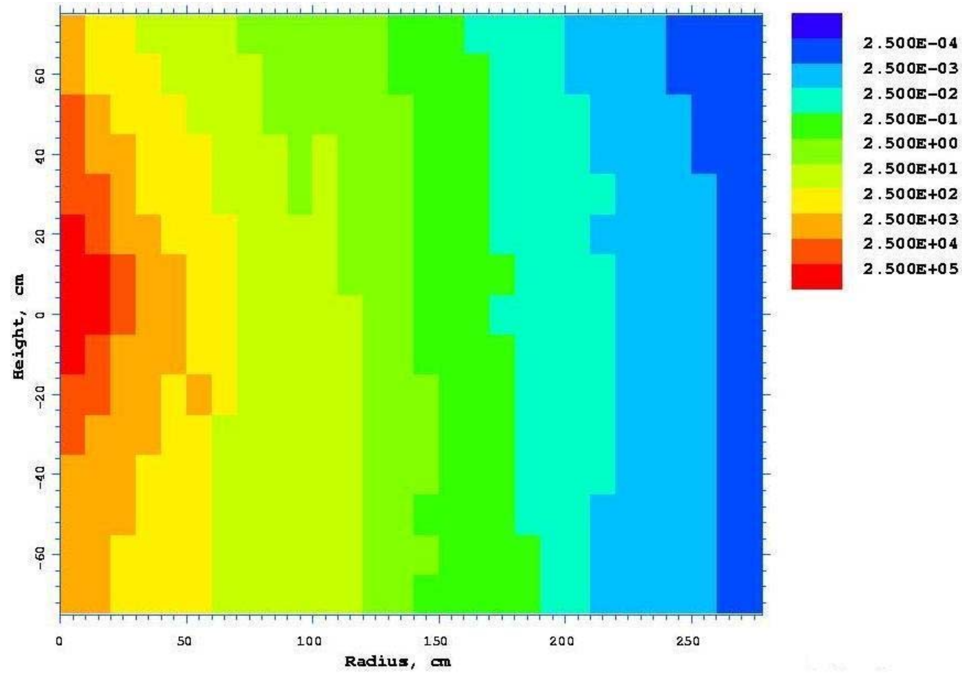


Figure 13. Neutron Dose map resulted from the use of 172-cm concrete shield, rem/hr

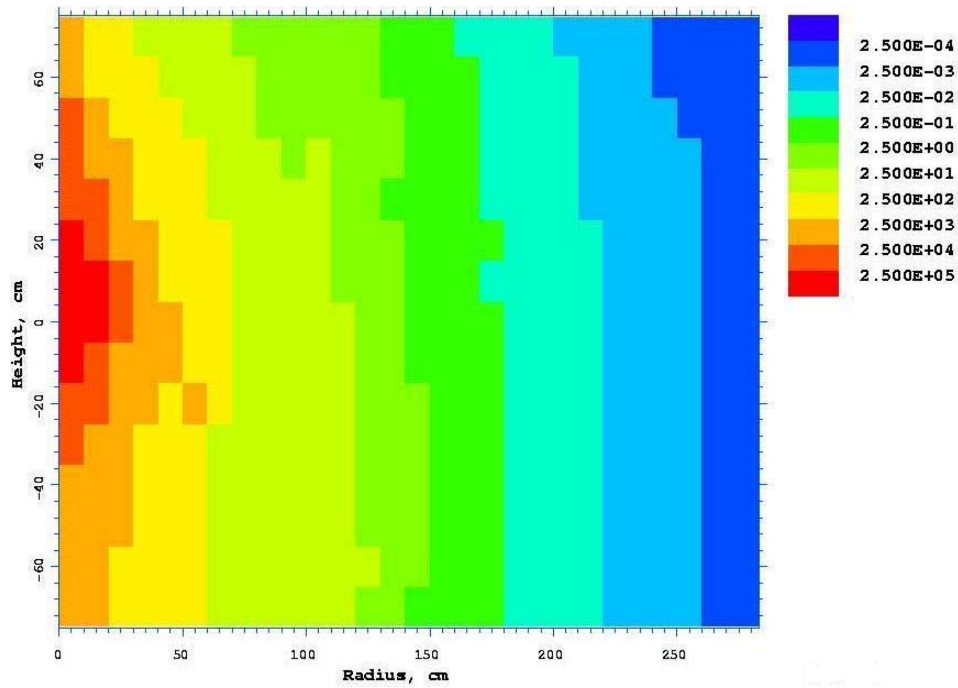


Figure 14. Neutron Dose map resulted from the use of 182-cm concrete shield, rem/hr

The final radial shield design with 182-cm of concrete was analyzed with different electron energy to obtain energy-dependent biological dose equivalent from neutron and photon outside the radial shield surface. The results are shown in Table V and Table VI. From the results, it can be observed that the 150 MeV electron beam case has a much smaller dose outside the radial shield relative to the 200 MeV electrons, while the 100 MeV electron beam case has an insignificant biological dose equivalent value. These results are consistent with the results and analysis from the simplified model discussed before.

Table V. Neutron biological dose equivalent outside the radial shield surface for 100 KW electron beam with electron energy of 100, 150, or 200 MeV

Upper energy limit (MeV)	100 MeV		150 MeV		200 MeV	
	Neutron dose (mrem/hr)	σ, %	Neutron dose (mrem/hr)	σ, %	Neutron dose (mrem/hr)	σ, %
1.00e-03	3.84e-05	12.1	7.81e-04	10.8	2.77e-03	10.6
1.00e-02	9.06e-06	11.9	1.87e-04	11.2	6.67e-04	10.6
1.00e-01	4.84e-05	12.2	1.00e-03	11.3	3.67e-03	10.7
5.00e-01	2.88e-04	12.0	5.99e-03	11.2	2.17e-02	10.7
1.00e+00	5.09e-04	11.9	1.04e-02	11.3	3.82e-02	10.7
2.00e+00	6.77e-04	12.3	1.45e-02	11.4	5.32e-02	10.8
5.00e+00	7.26e-04	12.2	1.68e-02	11.6	6.31e-02	11.1
1.00e+01	3.70e-04	12.7	8.18e-03	11.5	3.09e-02	11.1
2.00e+01	4.06e-04	12.1	7.55e-03	10.9	2.65e-02	10.4
2.00e+02	2.33e-03	13.2	6.07e-02	12.1	2.36e-01	11.4
Total	5.40e-03	12.6	1.26e-01	11.7	4.77e-01	11.1

Table VI. Photon biological dose equivalent outside the radial shield surface for 100 KW electron beam with electron energy of 100, 150, or 200 MeV

Upper energy limit (MeV)	100 MeV		150 MeV		200 MeV	
	Photon dose (mrem/hr)	σ, %	Photon dose (mrem/hr)	σ, %	Photon dose (mrem/hr)	σ, %
1.00e-01	2.32e-10	31.0	7.78e-09	15.0	3.45e-08	15.3
5.00e-01	6.17e-08	14.6	1.48e-06	11.4	5.27e-06	11.9
1.00e+00	1.02e-07	14.9	2.33e-06	11.5	8.50e-06	12.3
2.00e+00	2.58e-07	16.6	4.88e-06	12.0	1.67e-05	12.0
5.00e+00	3.96e-07	13.6	9.46e-06	11.3	3.44e-05	12.0
1.00e+01	2.37e-07	16.9	6.06e-06	11.9	2.27e-05	11.8
2.00e+01	1.50e-07	87.9	2.86e-07	23.3	1.32e-06	14.0
2.00e+02	0.0	0.0	0.0	0.0	0.0	0.0
Total	1.21e-06	16.1	2.45e-05	11.5	8.89e-05	11.9

III. Shielding Analysis of the Top Shield Section of the Subcritical Assembly

In the design of ADS, the shield above the subcritical assembly has more complicated geometry because of the electron beam tube and the required magnets for focusing and bending the electron beam. In addition, the electron beam tube has vacuum and the magnet support structure has void volume. The vacuum and the void volumes result in neutron streaming and require more complicated shielding analyses. In these analyses, detailed three dimensional calculational models for the MCNPX calculations were developed and used. The 0.5 mrem/hr biological dose equivalent limit, as explained before, is used to define the shield configuration in this area.

III.1 Results and Analysis Using a Simplified Geometrical Model of the Top Shield Section

Some preliminary analysis had been performed to determine the required shield thickness to achieve the acceptable dose limit above the shield. This preliminary analysis provided some initial shielding parameters, e.g., the thickness and shape of concrete (density of 4.8 g/cm^3) shield above the electron beam tube.

In the initial study, simplified cone shield geometry has been used to obtain the required dose level outside the shield. It is based on simple hand estimates and a geometrical model was sketched. Figure 15 show this initial sketch. To simplify the geometrical model, the complicated structure of the electron beam was neglected and simplified.

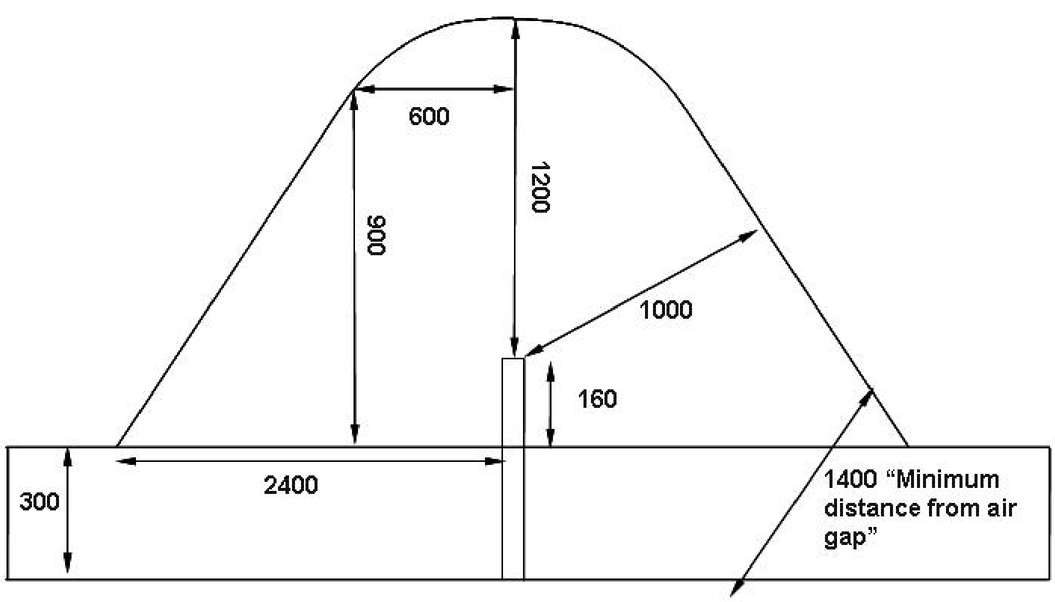


Figure 15 Simplified geometrical shielding sketch of the top section

The simplified geometrical shielding sketch of Figure 15 was used to develop a simple MCNPX model, which is shown in Figure 16. The shield has a 30 cm heavy concrete cylinder followed by a truncated cone, 240 cm in radius at the base, 136 cm height and 60 cm radius at the top. Due to the symmetry of this simplified geometrical model, two-dimensional R-Z mesh-based weight windows were used to generate the importance function. In this way, the spatial meshes are azimuthally homogenous and can be sampled adequately and the weight windows can be generated relatively easily. The weight windows generation was iterated and the converged weight windows are plotted in Figure 17 through 19. From the weight windows, it could be seen that the high energy neutrons generated in the target travel through the shield and contribute to the biological dose outside the shield surface. The low energy neutrons (< 10 MeV) generated in the fuel assemblies make very little contributions to the dose outside the shield surface.

The mesh based biological dose equivalent, which is calculated by the mesh tally capability of MCNPX and displayed in Figure 20. The results show that the biological dose is in the range of 0.1 to 1 mrem/hr. These results provide a rough estimate for the important parameters of the top shield.

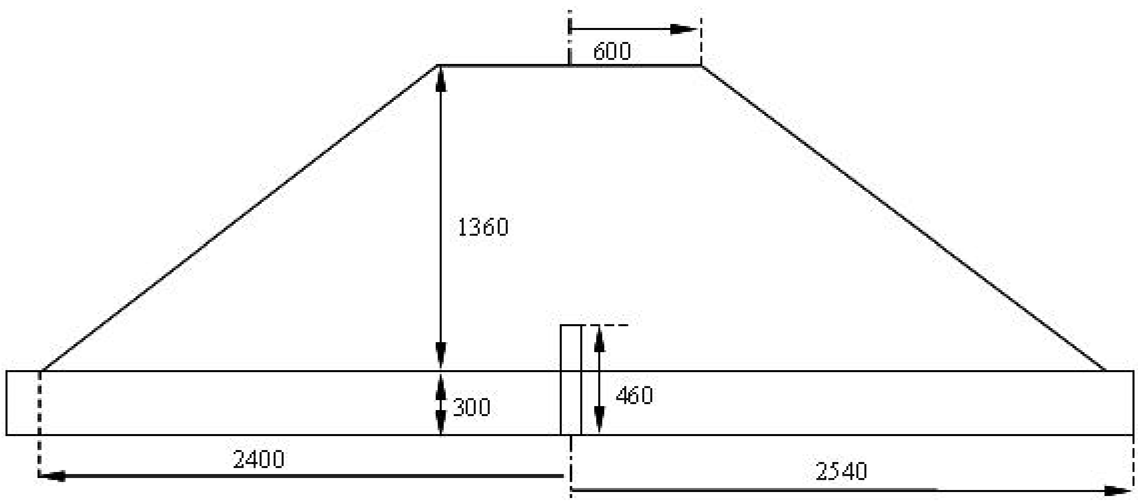


Figure 16. Dimension of the Beam Cover shield model. The model does not include the 50 cm top cover above the subcritical assembly

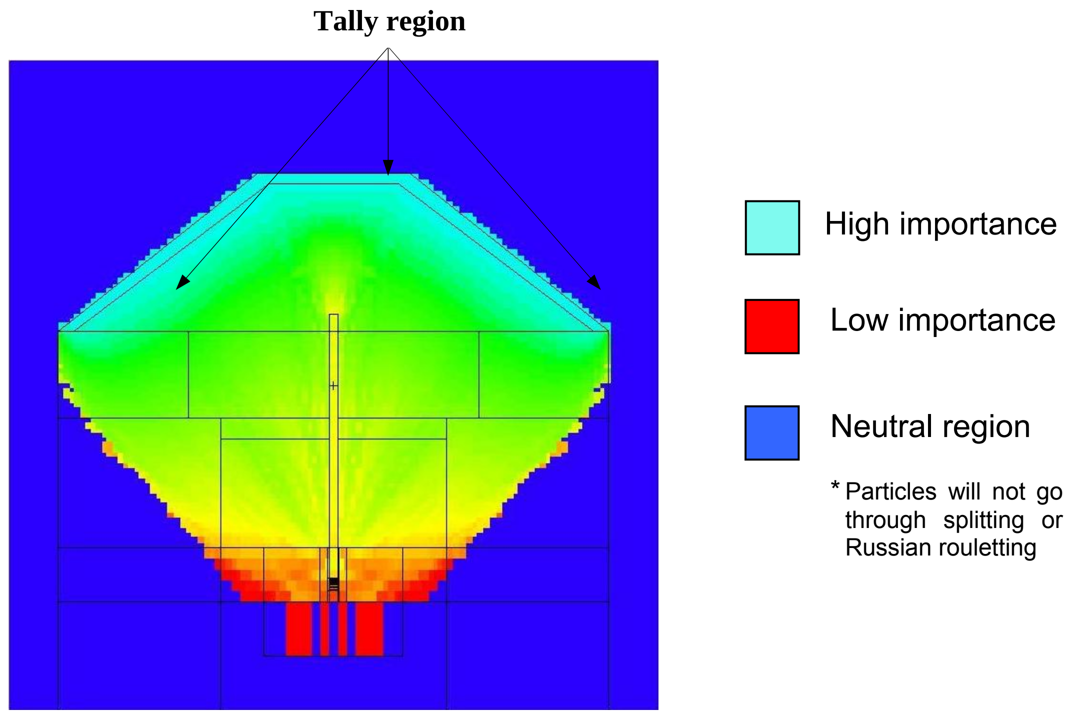


Figure 17. Weight windows of the shield top section for neutrons with energy >10 MeV

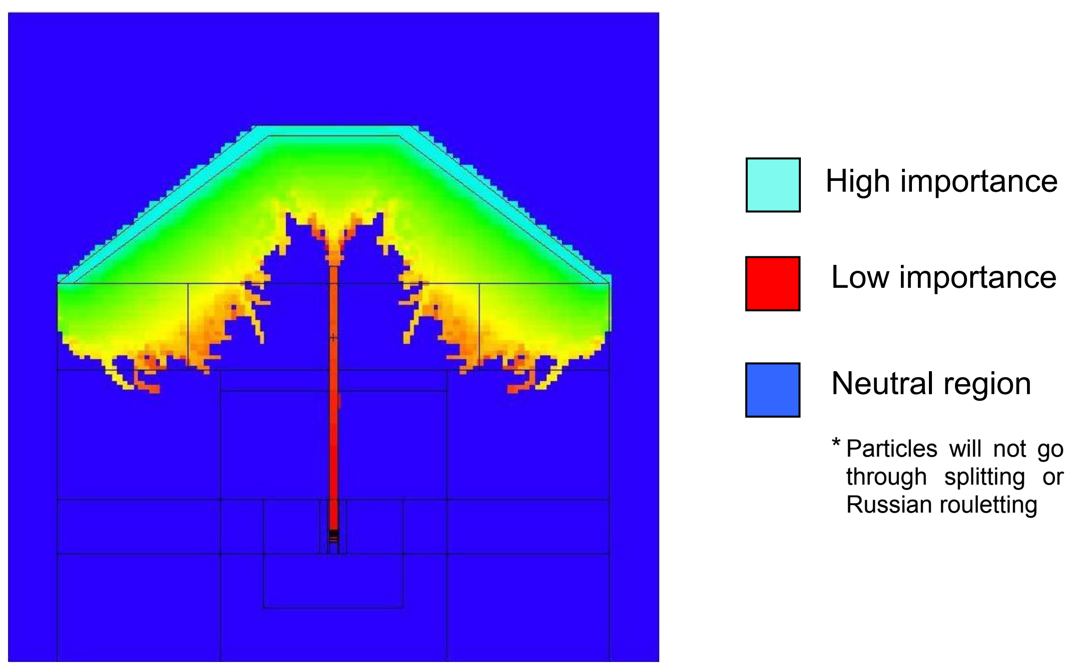


Figure 18. Weight windows of the shield top section for neutrons with energy between 0.1 MeV and 10 MeV

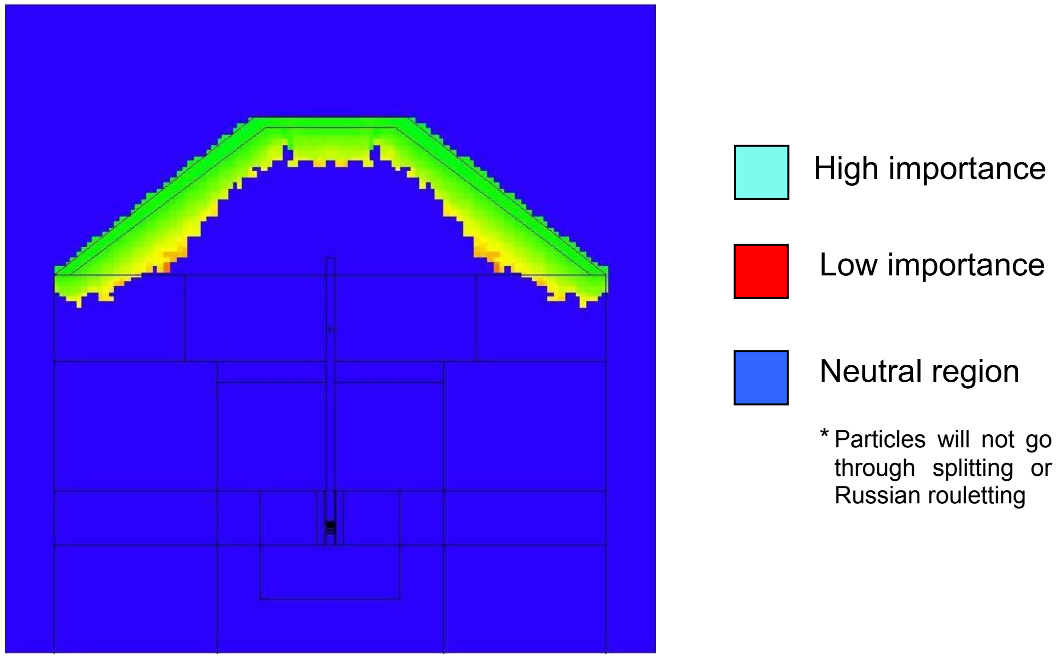


Figure 19. Weight windows of the shield top section for neutrons with energy < 0.1 MeV

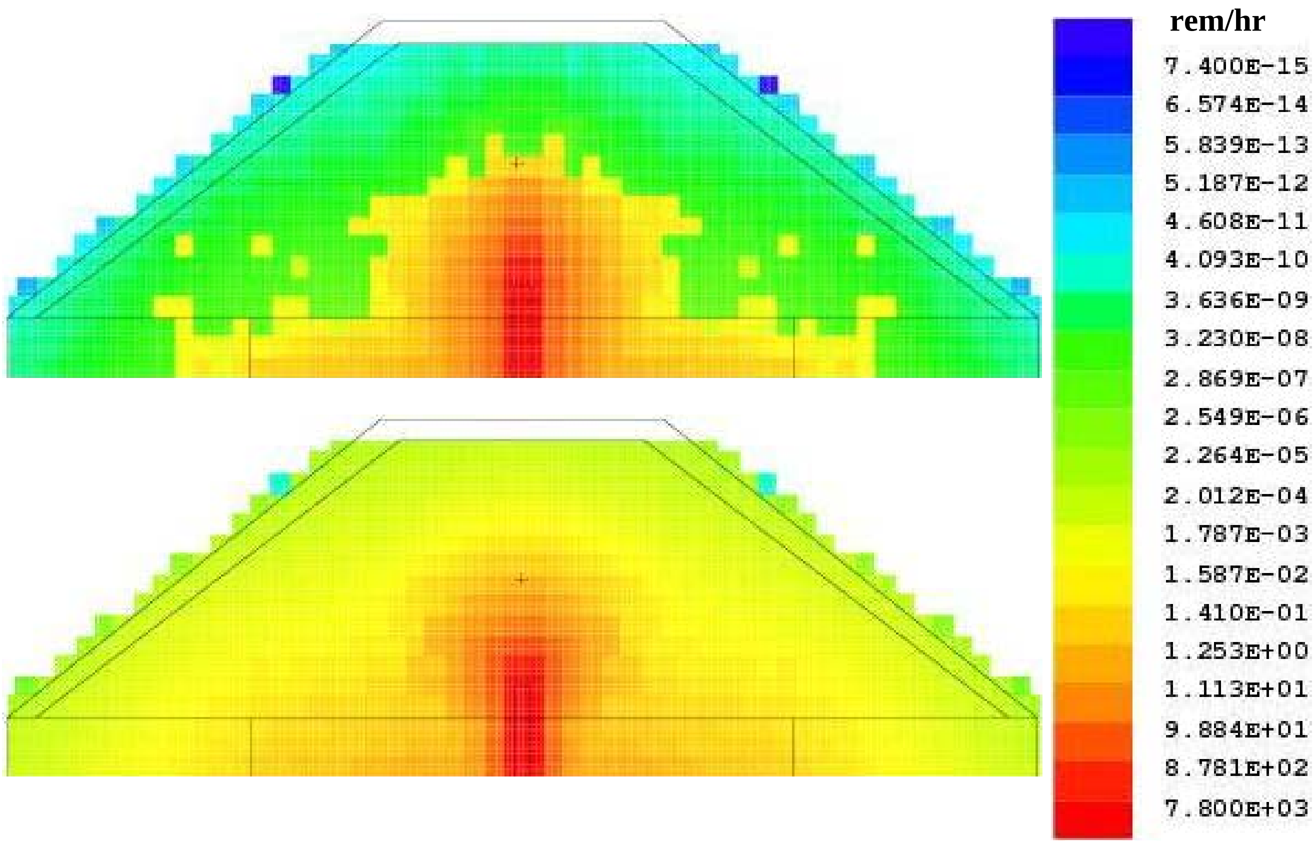


Figure 20. Mesh based photon (up) neutron (down) biological dose equivalent for the top shield section

III.2 Detailed Geometrical Model of the Top Shield Section

After the preliminary shield study was completed, the mechanical design was updated. Then, a three dimensional MCNPX calculational model has been developed. In this model, the electron beam, the magnet structure support, and the details of the electron beam channel are included. The cross section of the facility and the calculational model are shown in Figure 21. The main dimensions of the top shield section are given in Figure 22.

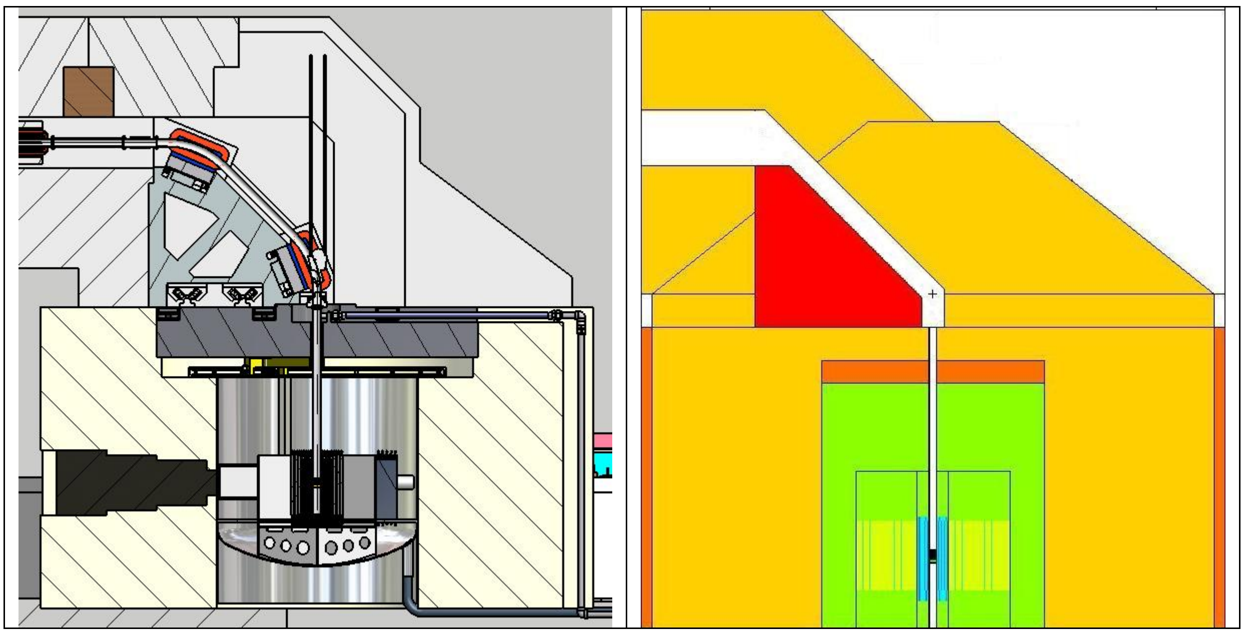


Figure 21. Vertical cross view (left part) and MCNPX geometrical model (right part)

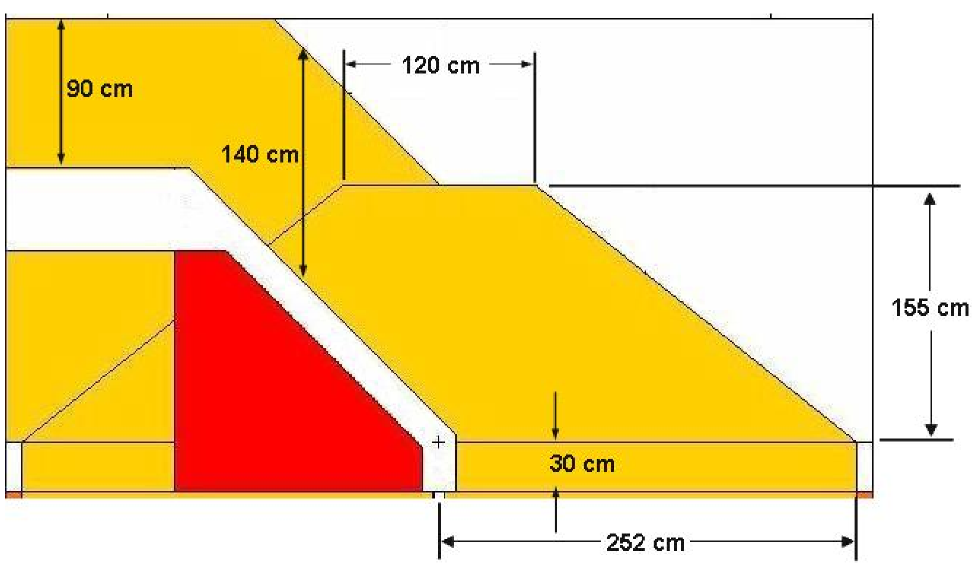


Figure 22. Important parameters for the top shield

III.3 Variance Reduction Technique

MCNPX and its variance reduction techniques are utilized for the analysis. The three-dimensional model shown in Figure 21 was used. It includes the electron beam, the target assembly, the fuel assemblies, the graphite reflector, the water tank, the radial concrete shield, and the top section details. For this model, three-dimensional (x-y-z) mesh is used for the weight windows generation because of the lack of symmetry. Thus, the weight windows shown in Figure 17 through 19 cannot be used for the new model and must be re-generated. In addition, the cell volume is relatively small since the cell dimensions is about one mean free path in each direction to avoid a large variation in the neutrons or photons weights inside the cell. The physics of the ADS facility show that the neutron yield is about 0.1 neutrons per electron, for 200 MeV electron beam. In addition, the high energy neutrons, E > 10 MeV, have a significant contribution to the biological dose equivalent outside the shield although they represent only a very small fraction of the source neutrons. Therefore, if electron source calculation is used directly, it is impossible to get sufficient number of neutron tracks in each mesh cell for calculating the importance function.

A different approach was used to reduce the required computational resources and to allow the neutron important functions to converge. The adopted approach can be summarized in the following three steps.

1 The MCNPX expensive electron source calculation with energy cut off is performed for generating a neutron surface source at the outer target surfaces. The MCNPX capability for generating a surface neutron source was used, which preserves the energy, the coordinates, and the direction of each neutron.
2 The neutron surface source is used in a series of iterative calculation for generating the neutron importance function using an iterative procedure with the same geometrical model.
3 After the neutron importance function is converged, the same geometrical model is used with the electron beam and without the surface source to calculate the biological dose outside the shield surface.

The weight windows generated in the second step for the dose tally are iterated to converge because of the statistical nature of the generator. To reduce the number of iterations, the shielding material densities are significantly reduced by a factor of 10^3 to 10^4 to insure that the neutron tracks covers the whole geometrical model. In other words, the attenuation factor of the shield is reduced to get neutrons everywhere in the geometrical model. Then, the densities of shielding materials are gradually increased during this iterative calculation for generating the weight windows until the densities of shielding materials reach the actual values. During these iterations, the expensive electron transport calculation is avoided, and it is relatively much faster to get sufficient neutron tracks for the three-dimensional mesh dose tally.

The high energy neutrons with E > 10 MeV, which are generated from the photonuclear reactions inside the target, are very important for the biological dose equivalent calculation outside the shield because of their large mean free path. Science their fraction is very small, the energy dependent weight windows were used to get an accurate estimate for their contribution. Weight windows with three energy groups were generated in the second step, and the final converged weight windows are shown in Figure 23 through 25. The mesh used for weight window is 10.0 cm, which is about one mean free path for the high energy neutrons. In these figures, the red color represents regions which make small contribution to the biological dose tally and they are assigned low importance. The light green color represents regions which make large contribution to the biological dose tally and they are assigned high importance. The blue color represents regions, which their lower weight bounds are zero, and they are named neutral region. In these zones, splitting and Russian roulette are not performed. Using the generated weight windows for the final dose calculation, the neutrons generated in the target with E > 10 MeV will keep on splitting when traveling toward and through the top shield. The neutrons with lower energy or neutrons traveling toward uninterested regions will vanish quickly. Therefore, the MCNPX calculation focus on the regions and neutron energy range contributing to the biological dose outside the top shield, which results in improved computational efficiency.

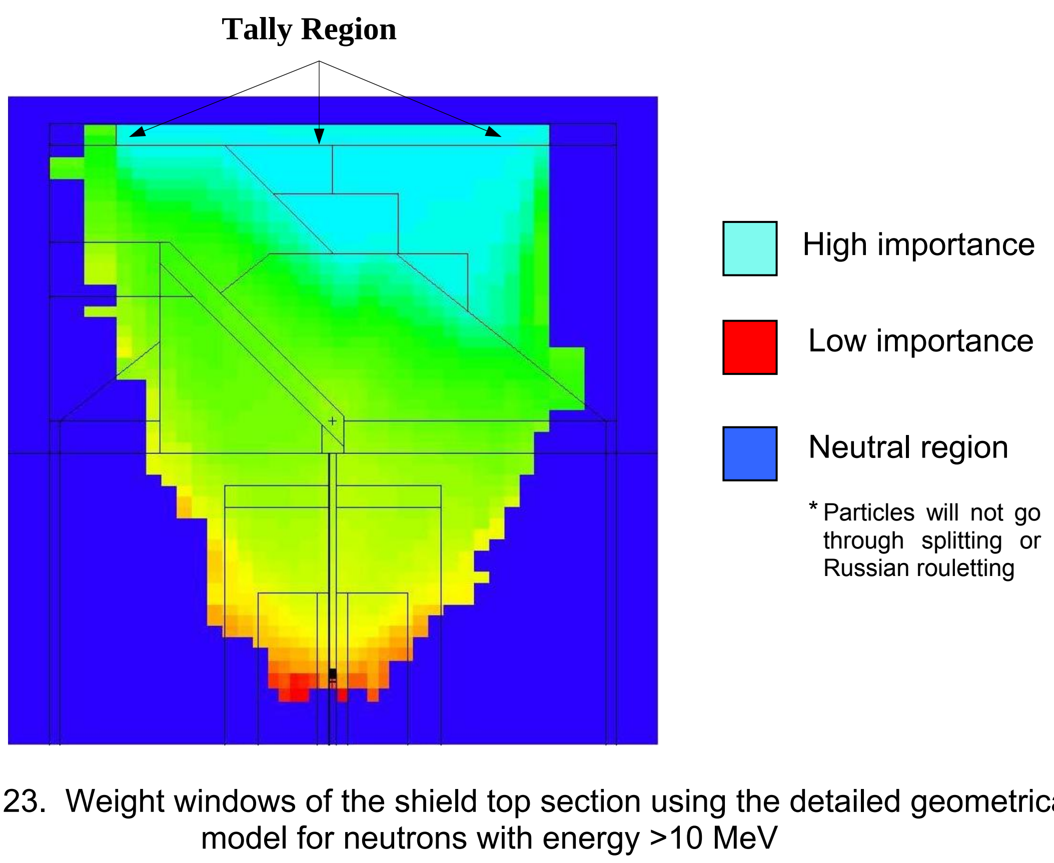


Figure 23. Weight windows of the shield top section using the detailed geometrical model for neutrons with energy >10 MeV

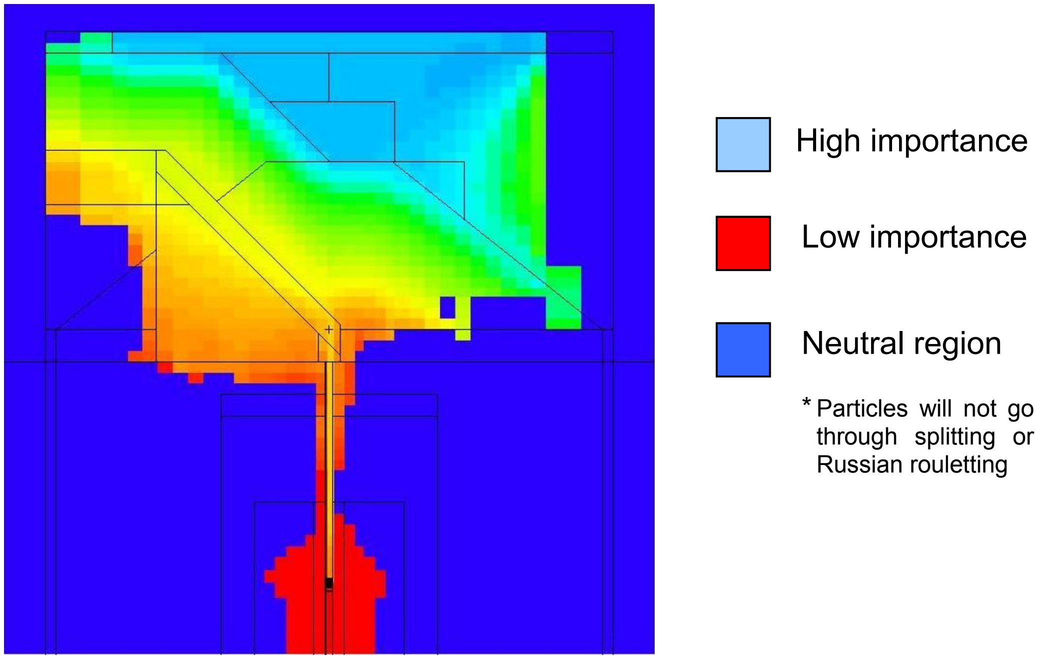


Figure 24 Weight windows of the shield top section using the detailed geometrical model for neutrons in the energy range of 0.1 MeV to 10 MeV

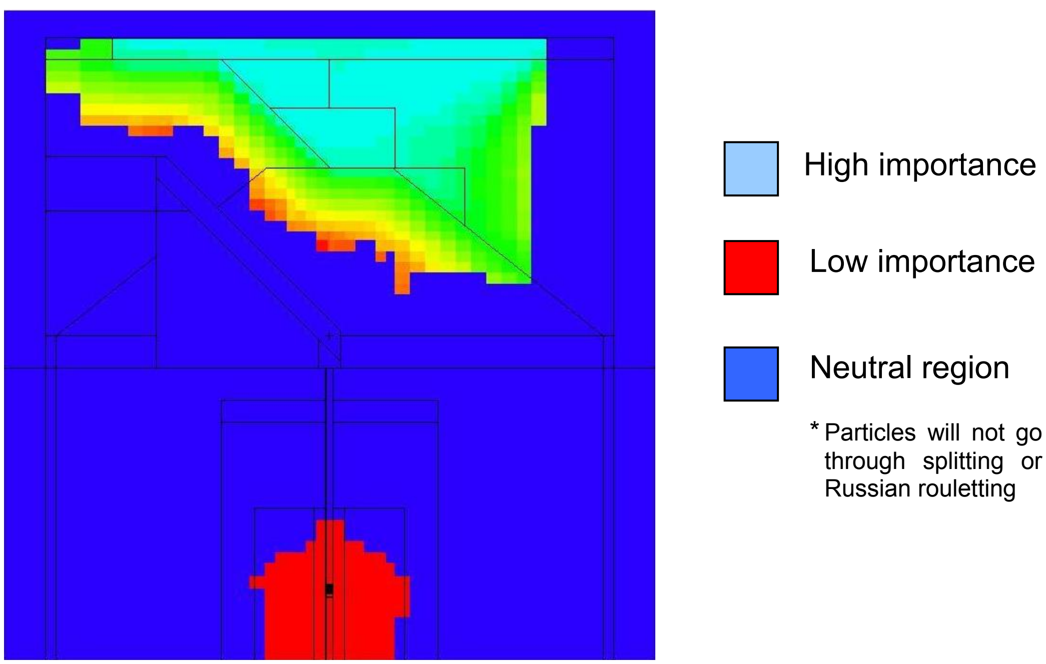


Figure 25. Weight windows of the shield top section using the detailed geometrical model for neutrons with energy >10 MeV

III.4 Calculated Dose Map Using the Detailed Geometrical Model

The mesh based dose calculated by the mesh tally capability of MCNPX using 200 MeV electron beam are depicted in Figure 26. From the results, it can be seen that at some positions are over shielded. Such over shielding is not desired since it increases the cost and the shield mass. These results were used to update the shield configuration for the final analysis.

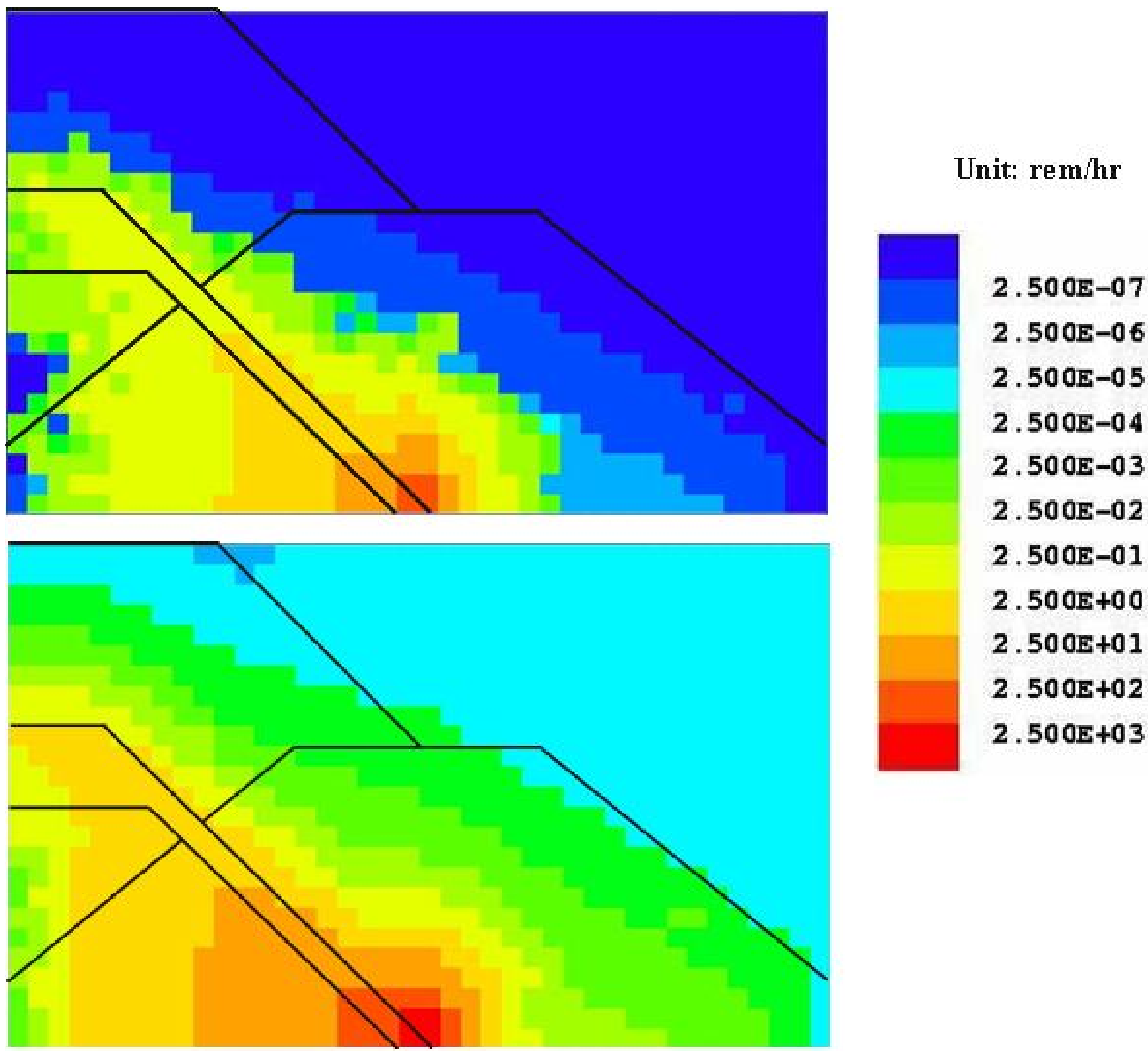


Figure 26. Biological dose maps from mesh tally calculation for photon (up) and neutron (down) for the top section of the shield

III.5 Analysis of the Updated Top Shield Section

Based on the results shown in Figure 26, the geometry of the top shield section has been updated and the new configuration is shown in Figure 27 through 30. In this way the biological dose equivalent everywhere outside the top shield is configured for a biological dose limit of 0.5 mrem/hr.

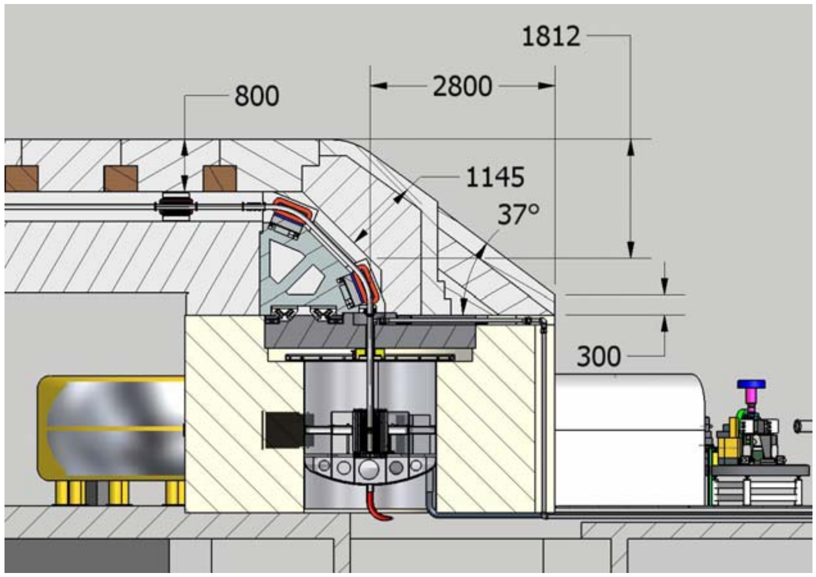


Figure 27. Updated vertical cross section of the biological shield configuration

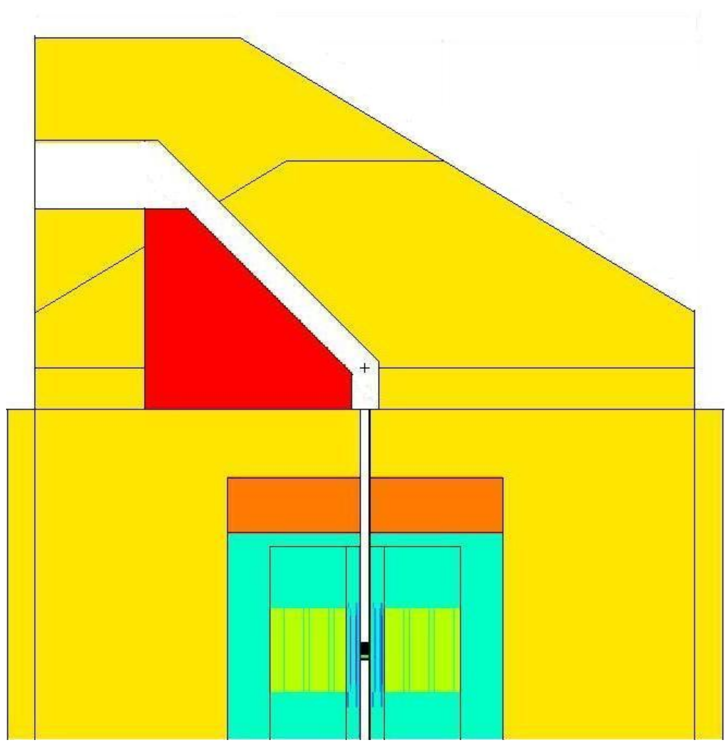


Figure 28. MCNP Model of the updated vertical cross section of the biological shield configuration

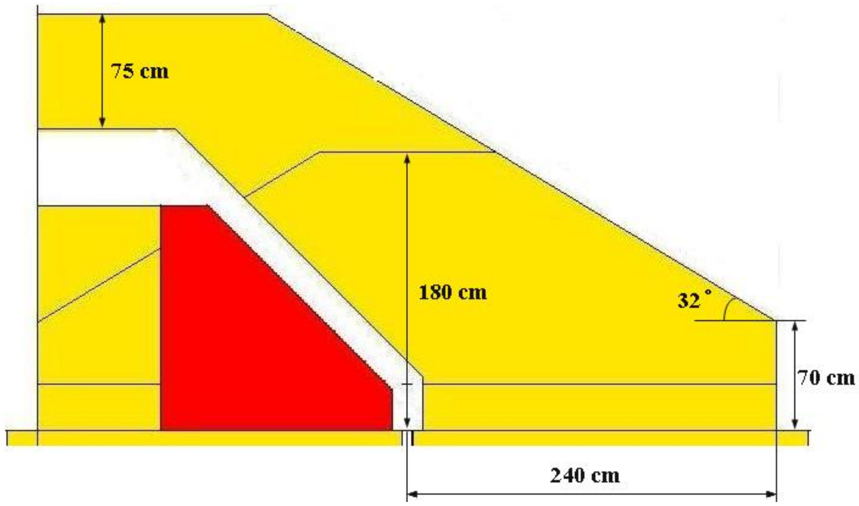


Figure 29. Dimensions of the updated vertical cross section of the biological shield configuration

The energy – dependent weight windows have been regenerated and the results are shown in Figure 30 through 32.

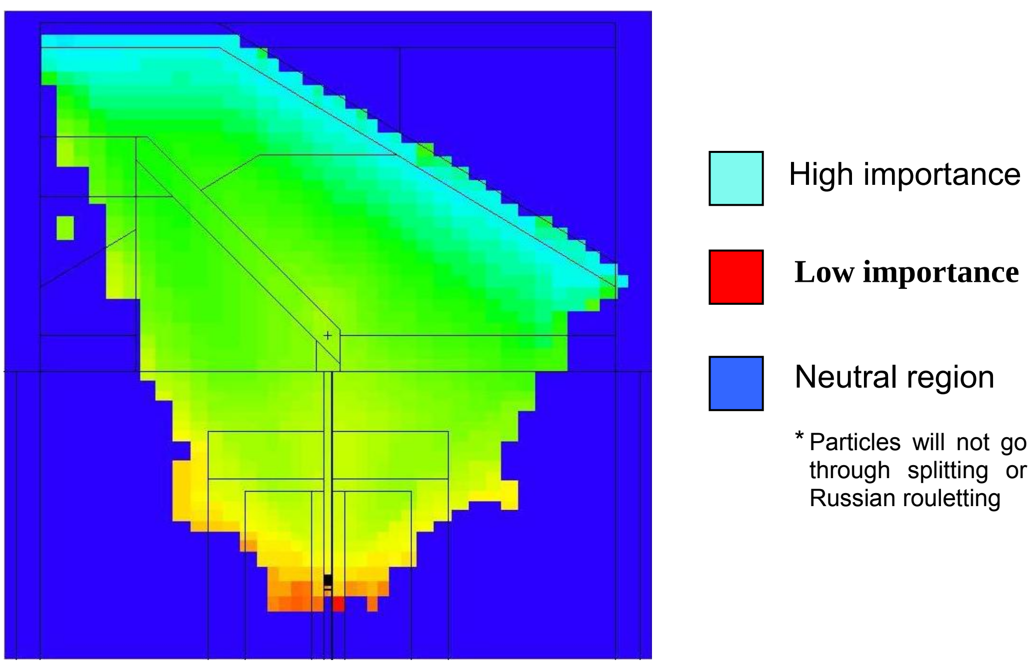


Figure 30. Weight Windows of the final design of top shield with electron beam channel for the neutrons with energy > 10 MeV

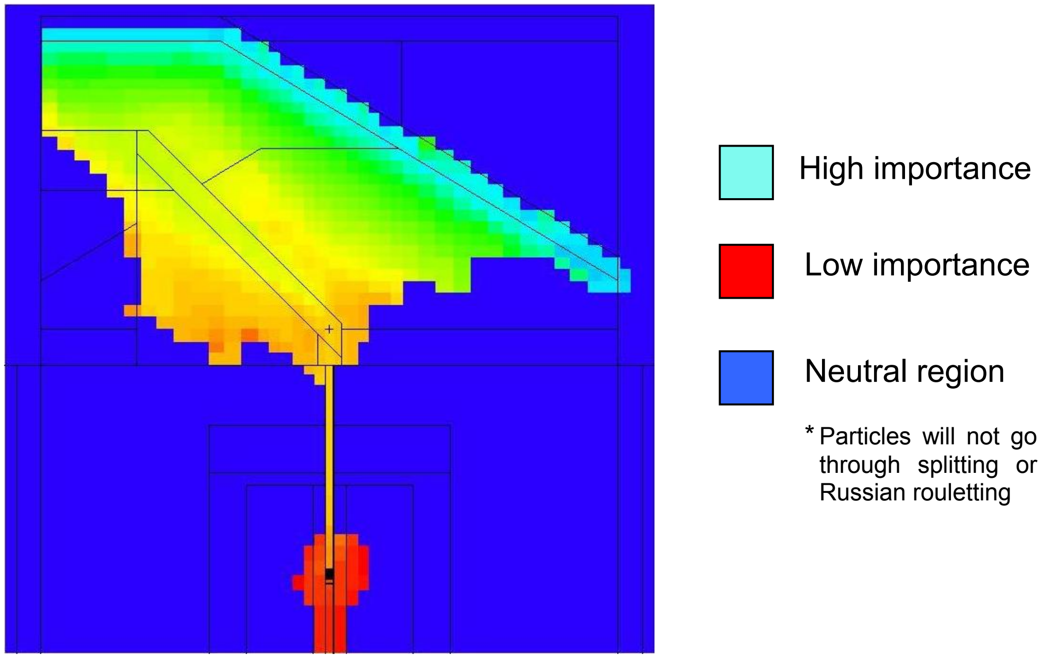


Figure 31. Weight Windows for the final design of top shield with electron beam for the neutrons with energy between 0.1 MeV and 10 MeV

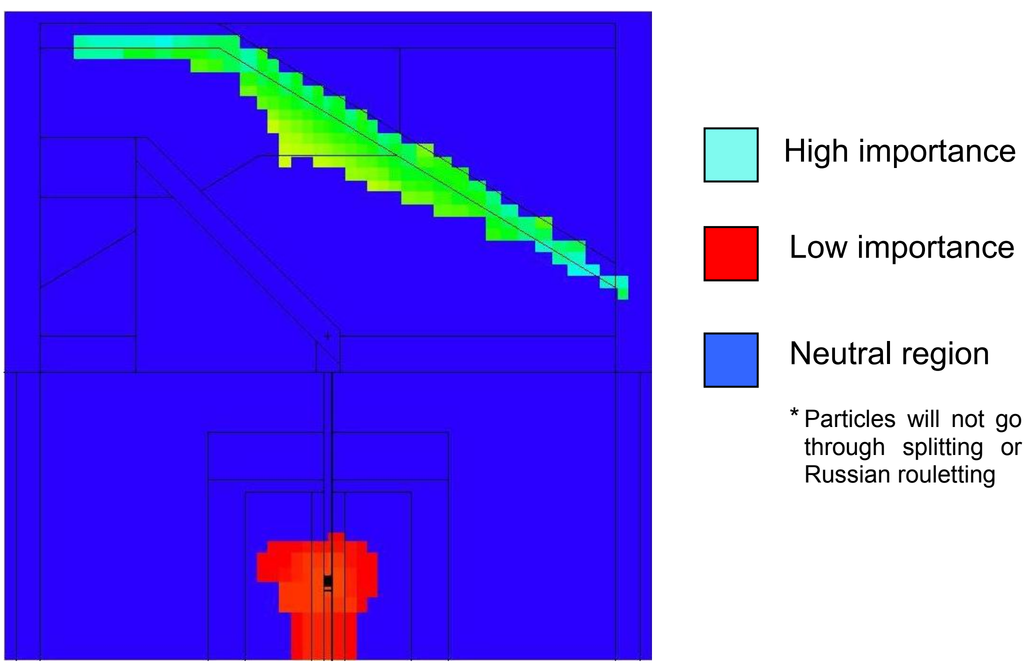


Figure 32. Weight Windows of the final design of top shield with electron beam for the neutrons with energy below 0.1 MeV

The biological dose equivalent maps due to the neutrons and the photons calculated by the mesh tally capability of MCNPX for the updated geometry are shown in Figure 33 and 34 respectively. It can be seen that the photon dose outside the shield is about four orders of magnitude less than the neutron contribution. The neutrons dominate the biological dose outside the top shield section. From Figure 33, it can be also seen that the 0.5 mrem/hr contour line is inside the top shield boundary, which satisfies the shielding design criterion. The statistical error of the calculation is less than 10%.

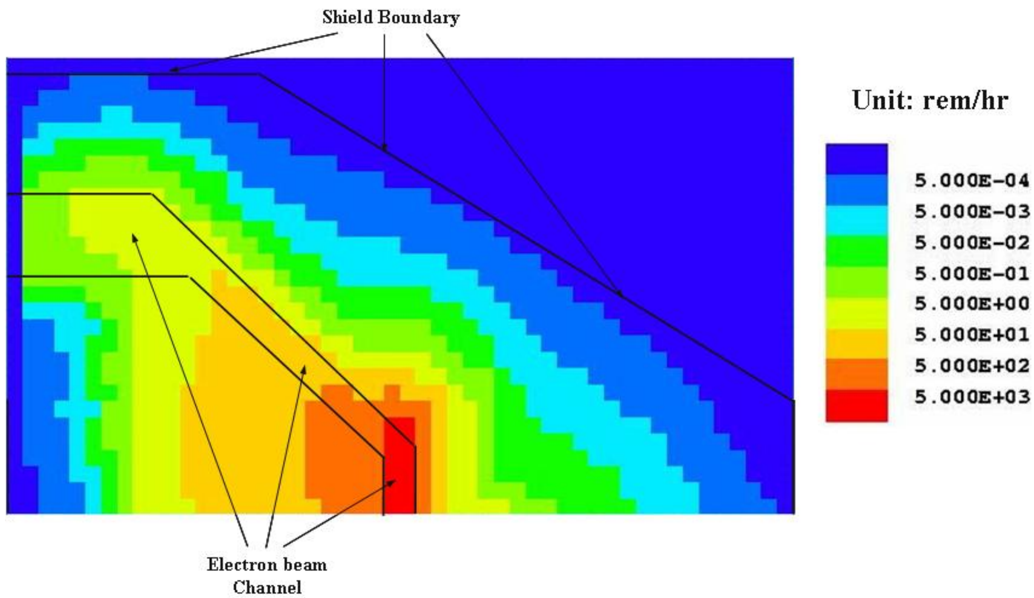


Figure 33. Neutron biological dose equivalent map for the top shield

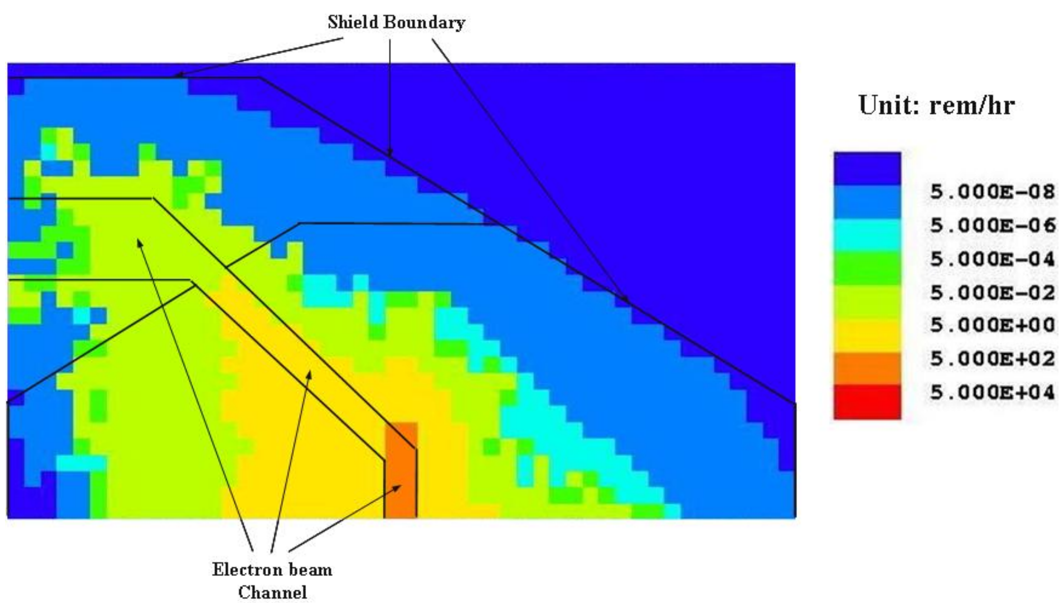


Figure 34 Photon biological dose equivalent map for the top shield

IV. Shielding Analysis of the Neutron Source Building

The building of the KIPT neutron source facility has an additional function, which is the protection of the general public according to the international limit. The shielding analysis was performed as a function of wall thickness. The building walls are made of normal concrete with a density 2.3 g/cm^3. The radial configuration of the facility is shown in Fig. 35. The facility hall size is 24 × 24 m, with the subcritical reactor located at the center.

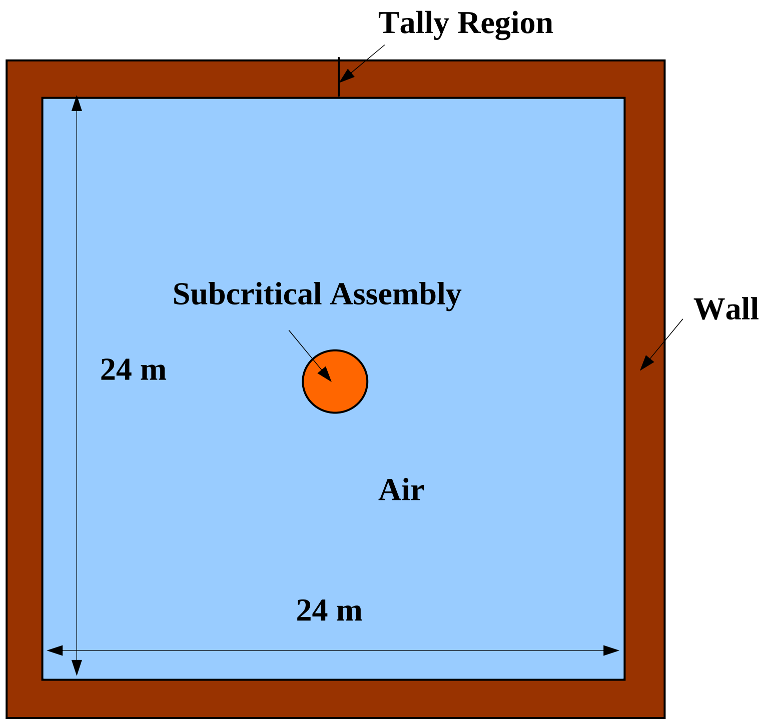


Figure 35 Radial Configuration of the Reactor Building

Based on ICRP for the general public, the radiation dose outside the reactor building should be less than than 0.0125-mrem/hr. The MCNPX code is used for this shielding study. Due to the facility size and the large attenuation factor, an analog MCNPX calculation is impractical, even if variance reduction techniques are used. To solve this problem, a two step method is introduced as follows:

1) The weight windows generated for defining the biological shield thickness of the subcritical assembly was used to calculate the neutron current on the outer radial surface of the heavy concrete shield, preserving the spatial, energy and angular distribution of the neutrons.
2) The generated neutron source was sampled in another MCNP calculation to obtain the biological dose equivalent as a function of the wall thickness.

The calculated biological dose equivalent from the neutron source is shown in Figure 36. From the results, it can be seen that a wall thickness greater than 10-cm of concrete is sufficient to attenuate the radiation dose for the general public.

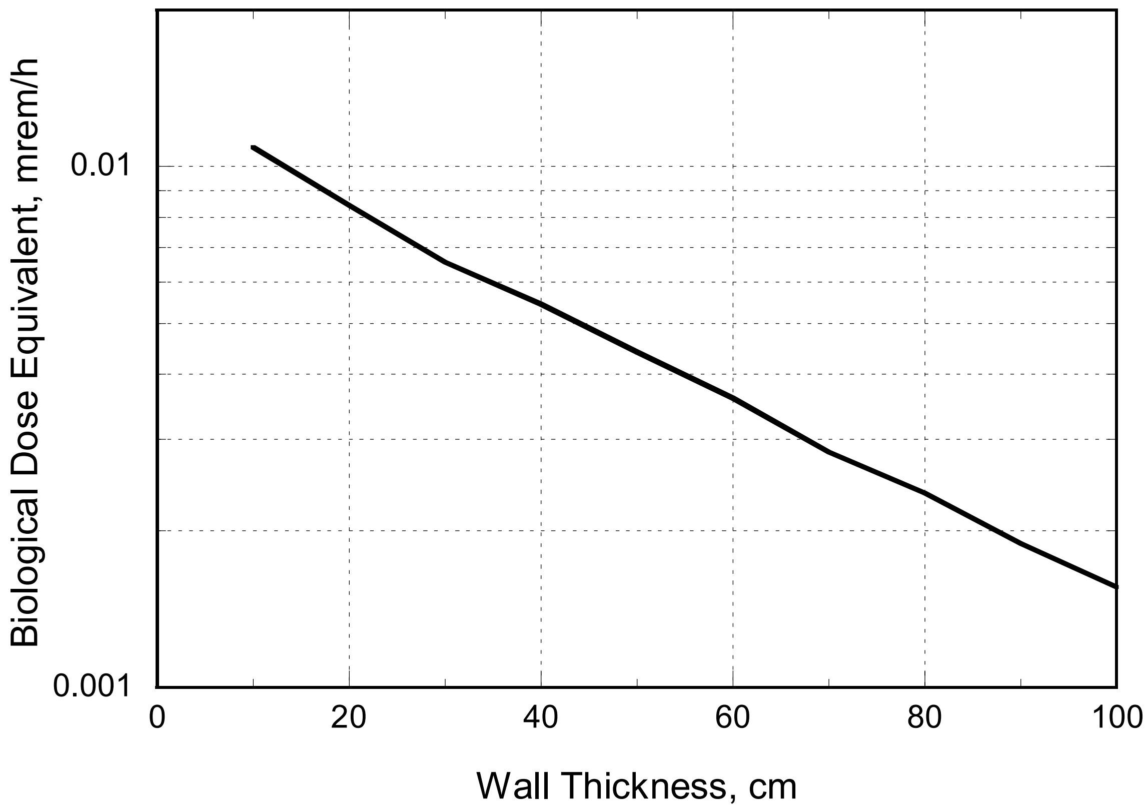


Figure 36. Biological Dose equivalent as function of the wall thickness of the Building

Conclusions

The shielding analyses of the accelerator driven subcritical system (ADS) was performed successfully using the Monte Carlo Code MCNPX and the variance reduction technique based on the weight windows generation. An efficient procedure for generating the neutron weight windows has been developed and used successfully. The expensive electron transport calculation was avoided during the weight windows iterations, which improved significantly the computational efficiency. In the complicated shielding problems of ADS, this methodology can get the results with sufficient statistical error of less than 10% and it makes the three-dimensional mesh based weight windows generation practical to use. The developed methodology was used successfully to define the shielding configuration, which satisfies the design criterion and reduce the total cost as much as possible. The cost reduction was obtained by eliminating the unnecessary shielding materials from the complicated geometry.

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