Investigation of Various Shielding Methods for an Isotropic 14.1 MeV Neutron Source

Zachary D. Whetstone, **Adrienne L. Lehnert, Tomasz Zak, Kimberlee J. Kearfott**

University of Michigan, Department of Nuclear Engineering and Radiological Sciences

Introduction

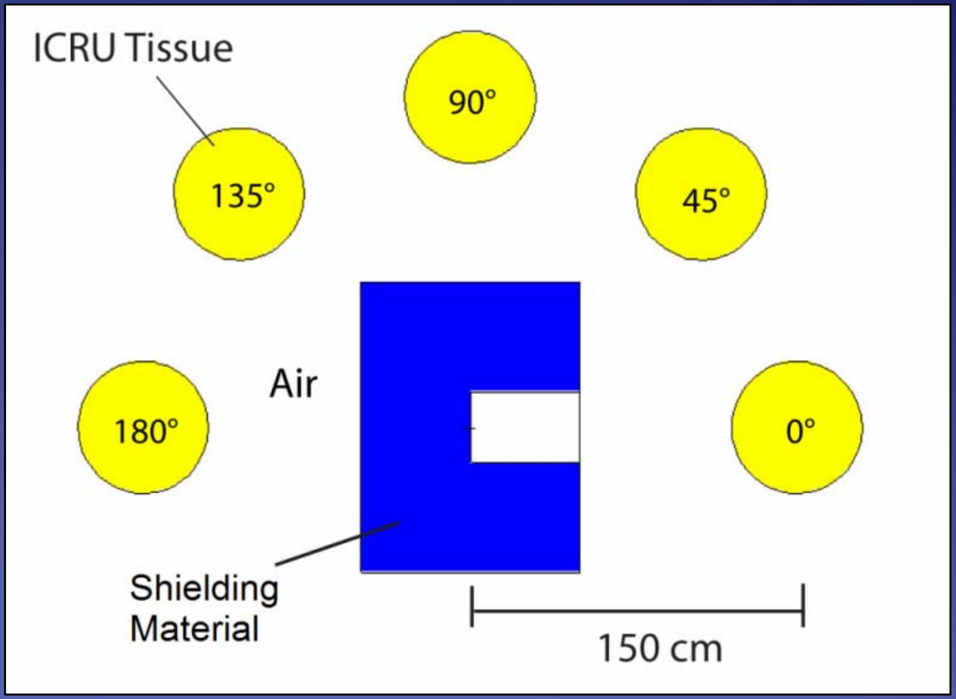
- Want to design “portable” shielding for D,T neutron generator
- Source is to be used as a means of determining the lower Z component of material composition within luggage and shipping containers
- Shielding could be applied to any application where it is necessary to limit the angle of incident neutrons

Materials and Methods

- Monte Carlo N-Particle Transport Code, Version 5 (MCNP 5) was used to simulate the interaction of neutrons in various materials around an isotropic 14.1 MeV neutron source
- Materials tested included :
  - 1080 Steel
  - Lead
  - Tungsten
  - Polyethylene
  - Borated polyethylene
- Shielding material was arranged in a cylinder around the source, with one end open to allow the neutrons to emerge unattenuated

Materials and Methods

- Neutron energy flux through various spheres was estimated with a surface current tally (F1) in MCNP5
- Spheres filled with ICRU four-component tissue
- Neutron and gamma ray fluence obtained from F4 tally and multiplied by dose conversion factors found in ICRP Publication 51 and ICRU Report 47 to obtain dose

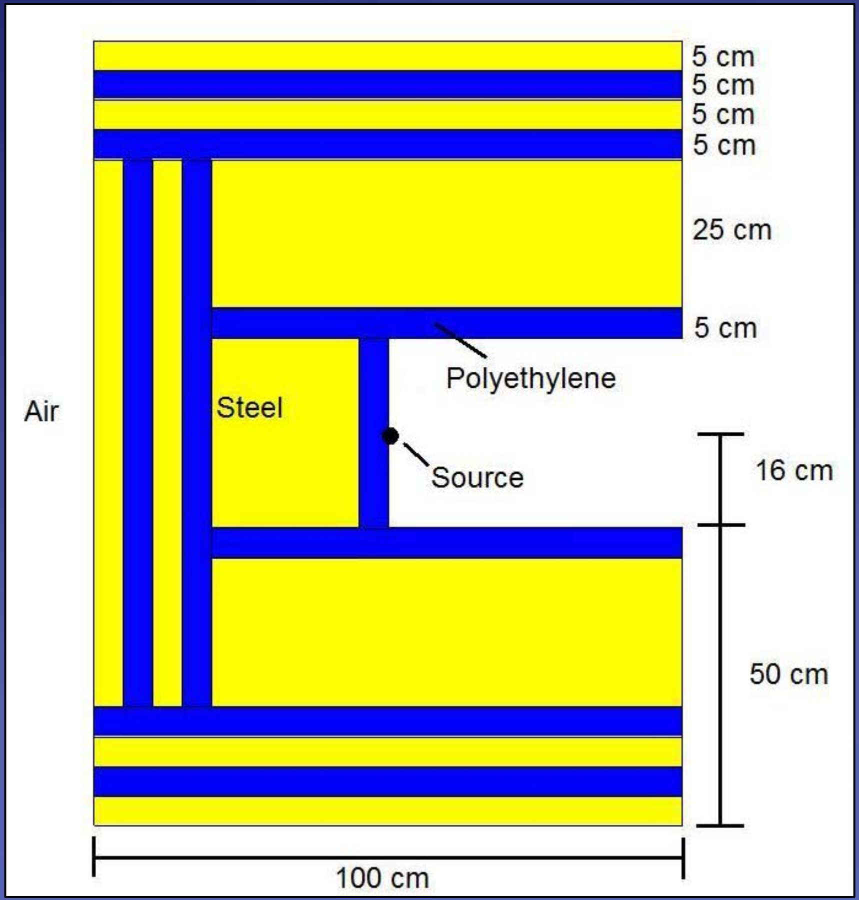


Shielding Configurations

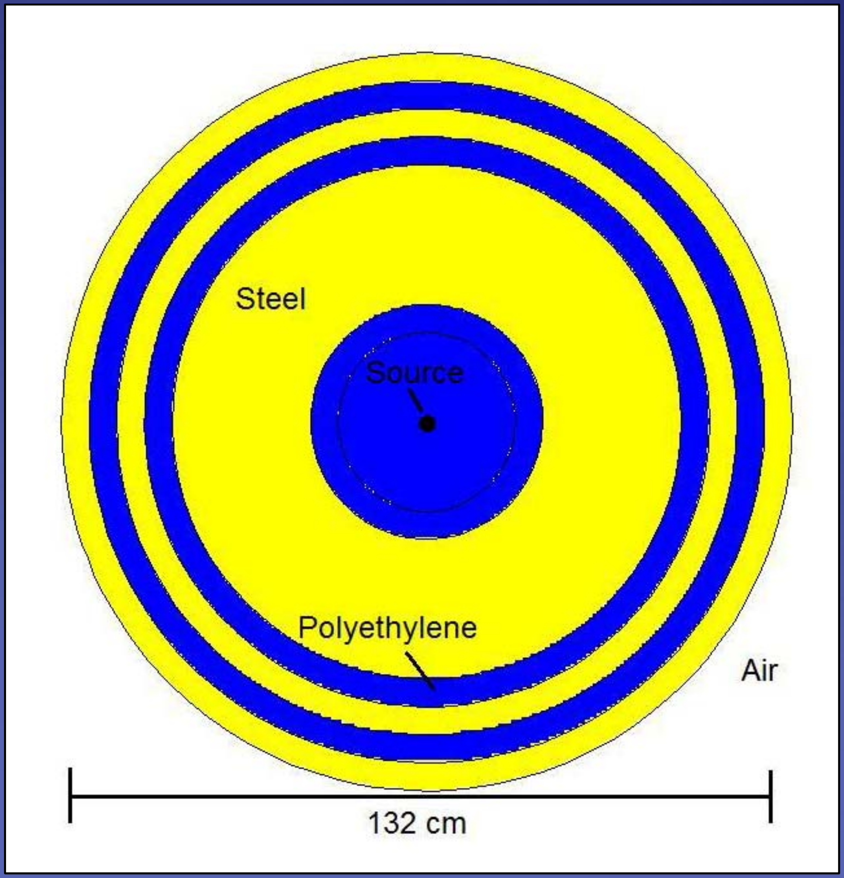
Geometry	Description
0	Unshielded source
1	50 .0 cm of polyethylene surrounding the source
2	6.0 cm of 1080 steel surrounding the source
3	Alternating layers of steel and polyethylene: 5.0 cm polyethylene, 25.0 cm steel, 5.0 cm polyethylene, 5.0 cm steel, 5.0 cm polyethylene, 5.0 cm steel
4	Geometry 3 with an additional 5.0 cm of lead on the outside
5	Geometry 3 with the outside layer of steel removed
6	Geometry 3 with outside layer of steel and polyethylene removed
7	A layer 15.0 cm thick of polyethylene around the source surrounded by 35.0 cm thick of steel
8	A layer 35.0 cm thick of steel around the source surrounded by 15.0 cm thick of polyethylene
9	Geometry 3 with the polyethylene doped with 3% ^{10}B by weight
10	Geometry 3 with the polyethylene doped with 10% ^{10}B by weight
11	Alternating layers of tungsten and polyethylene: 5.0 cm polyethylene, 10.2 cm tungsten, 5.0 cm polyethylene, 2.0 cm tungsten, 5.0 cm polyethylene, 2.0 cm tungsten

Shielding Configurations

Side view of layered shielding configuration



Front view of layered shielding configuration

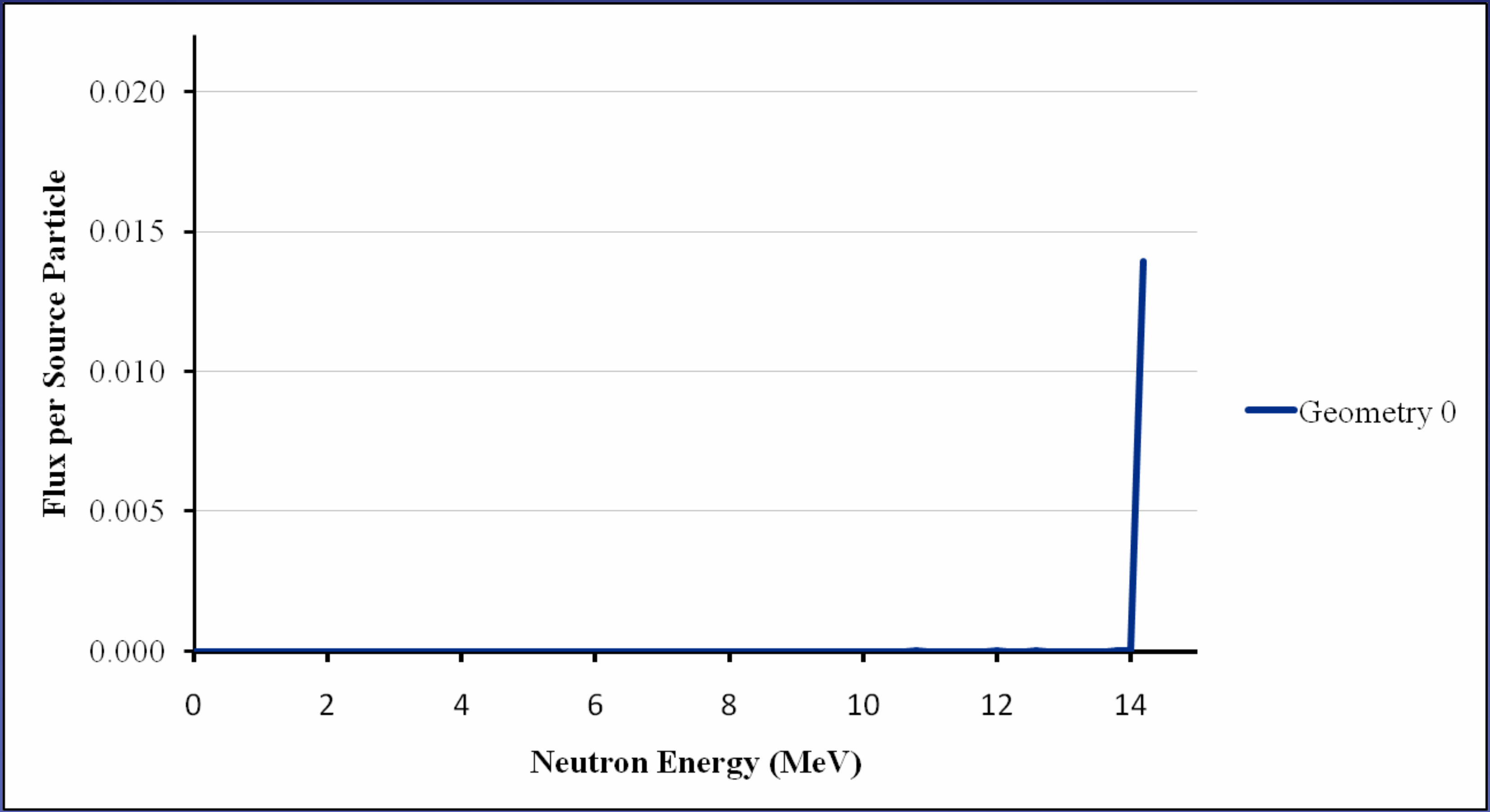


Activation Concerns

- Steel will deflect some neutrons towards target or through moderating material
- Reflected neutrons will maintain most of their original energy due to large mass of nuclei
- Steel will become activated
- Tungsten can substitute for steel without activation
- High price of tungsten makes steel a more realistic option

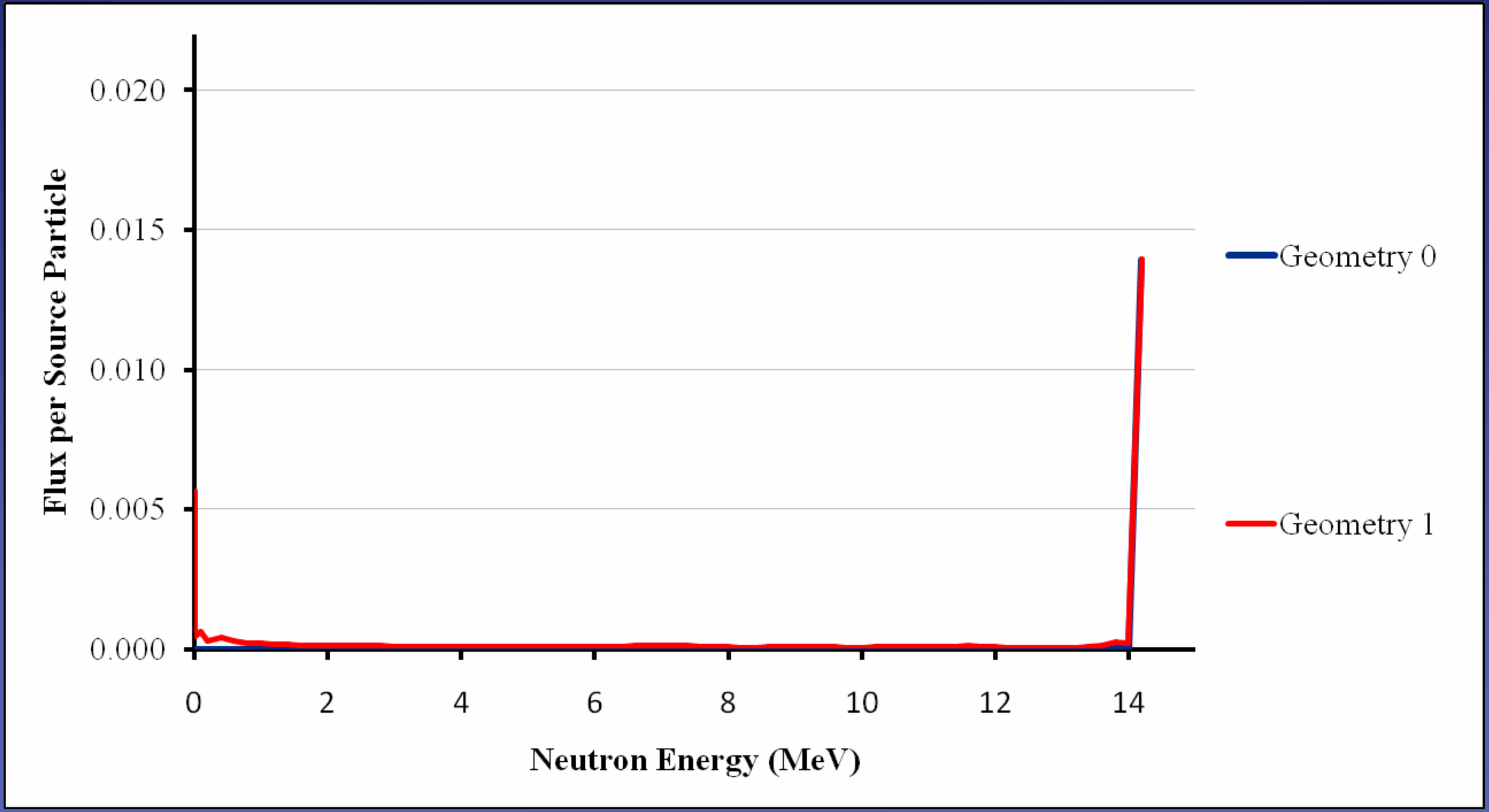
Results

The neutron energy spectra for an unshielded source



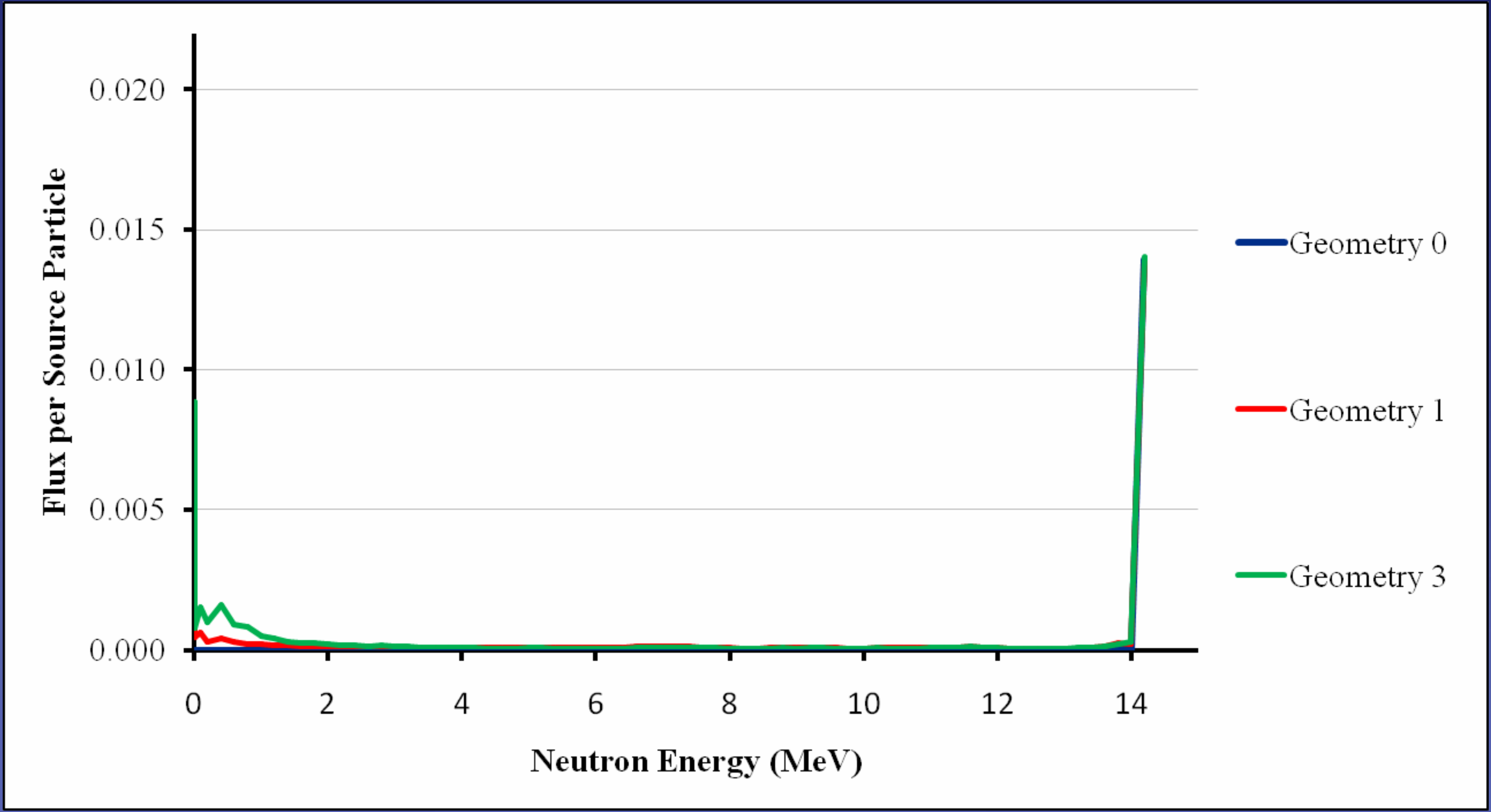
Results

The neutron energy spectra through the sphere at 0° for a source shielded by 50 cm of polyethylene



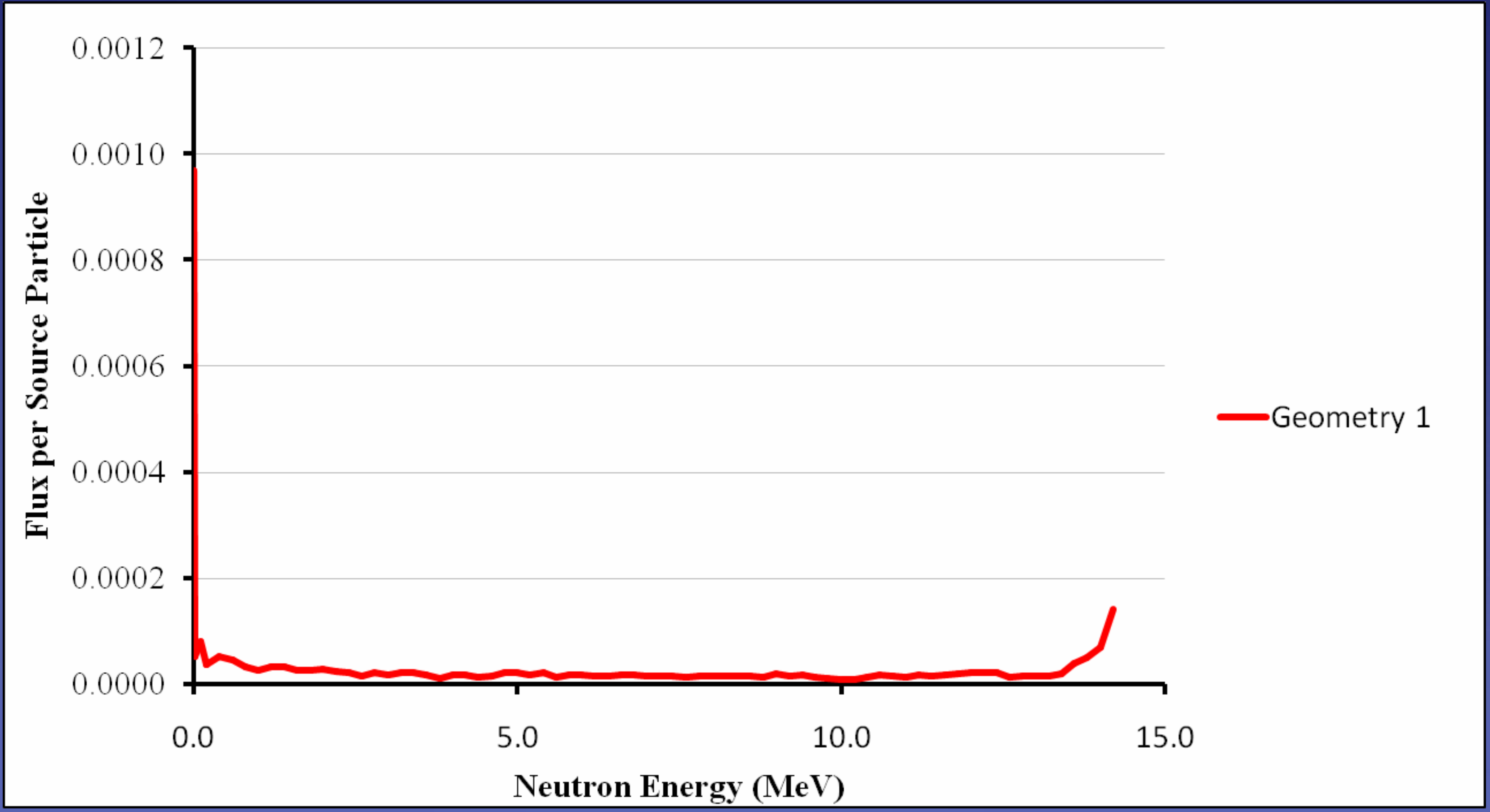
Results

The neutron energy spectra through the sphere at 0° for a source shielded by alternating layers of steel and polyethylene



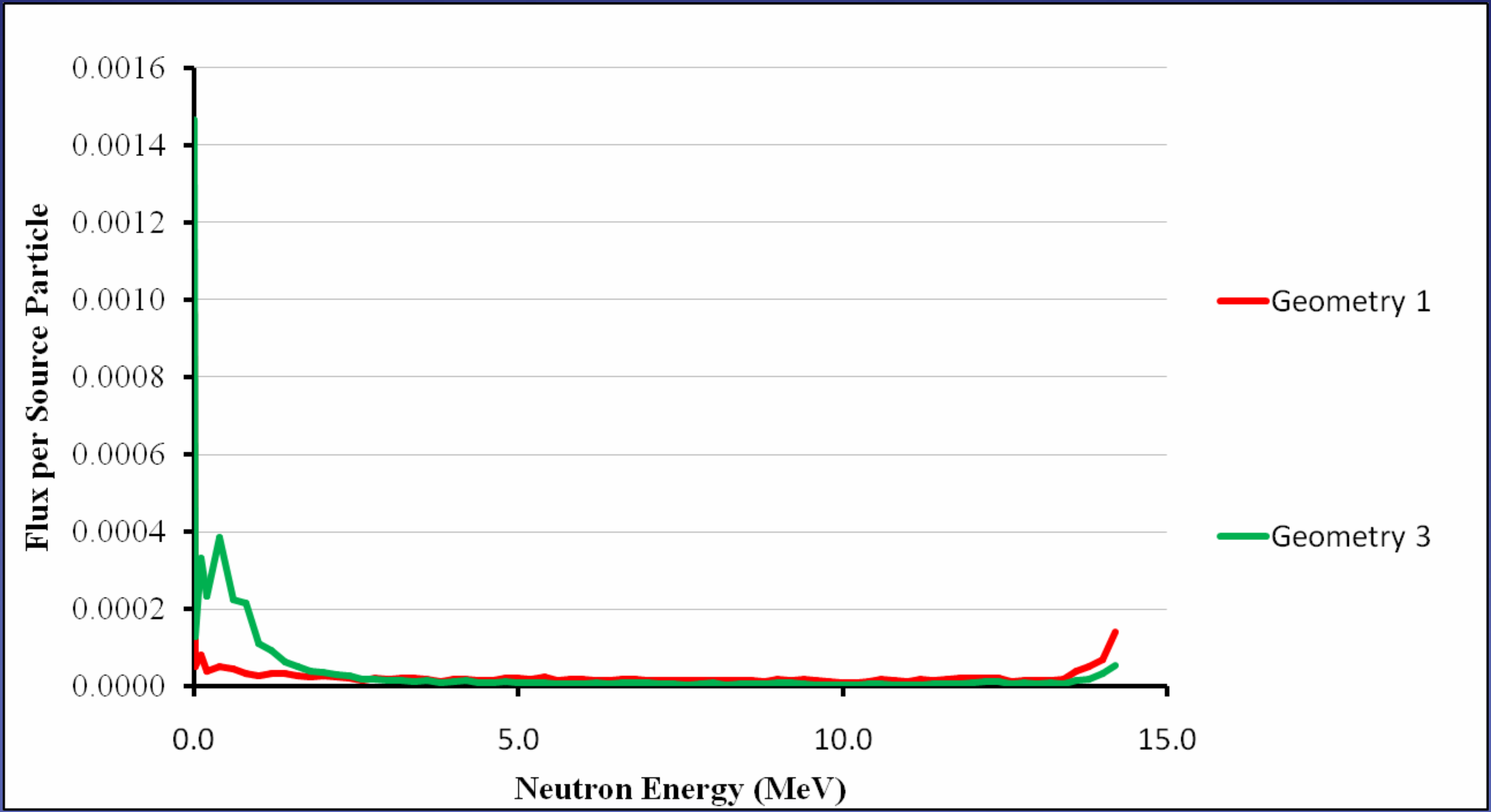
Results

The neutron energy spectra through the sphere at 45° for a source shielded by 50 cm of polyethylene



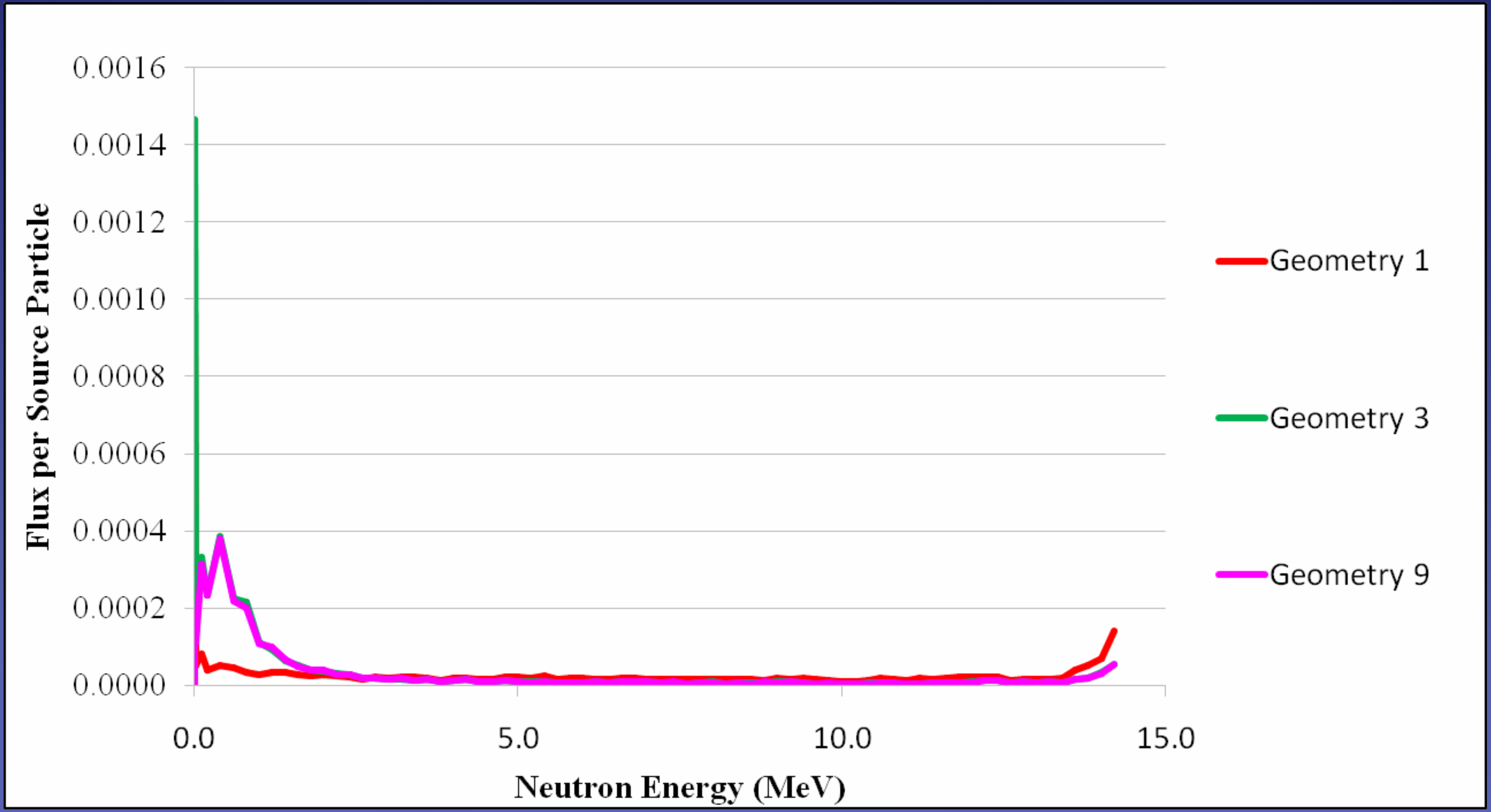
Results

The neutron energy spectra through the sphere at 45° for a source shielded by alternating layers of steel and polyethylene



Results

The neutron energy spectra through the sphere at 45° for a source shielded by alternating layers of steel and borated polyethylene



Dose Equivalent to ICRU Sphere

		Geometry											
		0	1	2	3	4	5	6	7	8	9	10	11
0°	Neutron Dose	213	275	294	281	281	281	282	277	337	279	281	272
	Photon Dose	1.27	3.24	3.79	4.10	3.62	4.06	4.05	3.50	4.95	3.01	3.00	3.47
	Total Dose	215	278	298	285	284	285	286	281	342	282	284	275
45°	Neutron Dose	213	16.9	109	11.0	9.89	14.9	20.6	13.6	20.7	10.5	11.0	40.2
	Photon Dose	1.28	0.934	1.47	0.532	0.270	0.701	0.826	0.54	1.01	0.309	0.310	0.902
	Total Dose	214	17.9	111	11.5	10.2	15.6	21.4	14.1	21.7	10.9	11.3	41.1
90°	Neutron Dose	212	6.10	94.1	1.14	0.869	1.73	3.04	4.25	1.19	1.08	1.18	6.70
	Photon Dose	1.28	0.663	1.33	0.116	0.0372	0.246	0.239	0.116	0.838	0.0375	0.0388	0.128
	Total Dose	214	6.76	95.4	1.25	0.906	1.98	3.28	4.36	2.03	1.12	1.22	6.82
135°	Neutron Dose	214	2.42	124	0.453	0.305	0.772	1.78	2.87	0.351	0.431	0.485	4.49
	Photon Dose	1.29	0.483	1.89	0.0506	0.00765	0.167	0.180	0.0856	0.755	0.0131	0.0142	0.107
	Total Dose	215	2.90	126	0.504	0.313	0.939	1.96	2.96	1.11	0.445	0.499	4.60
180°	Neutron Dose	213	8.10	154	1.32	0.884	2.65	4.55	5.82	1.11	1.27	1.43	10.6
	Photon Dose	1.27	0.788	2.23	0.140	0.0219	0.336	0.361	0.137	0.859	0.0315	0.034	0.213
	Total Dose	215	8.89	157	1.46	0.906	2.98	4.91	5.96	1.97	1.30	1.46	10.8

Value in table is multiplied by 1×10^{-17} Sv per source neutron

Dose Equivalent to ICRU Sphere

		Geometry											
		0	1	2	3	4	5	6	7	8	9	10	11
0°	Neutron Dose	213	275	294	281	281	281	282	277	337	279	281	272
	Photon Dose	1.27	3.24	3.79	4.10	3.62	4.06	4.05	3.50	4.95	3.01	3.00	3.47
	Total Dose	215	278	298	285	284	285	286	281	342	282	284	275
45°	Neutron Dose	213	16.9	109	11.0	9.89	14.9	20.6	13.6	20.7	10.5	11.0	40.2
	Photon Dose	1.28	0.934	1.47	0.532	0.270	0.701	0.826	0.54	1.01	0.309	0.310	0.902
	Total Dose	214	17.9	111	11.5	10.2	15.6	21.4	14.1	21.7	10.9	11.3	41.1
90°	Neutron Dose	212	6.10	94.1	1.14	0.869	1.73	3.04	4.25	1.19	1.08	1.18	6.70
	Photon Dose	1.28	0.663	1.33	0.116	0.0372	0.246	0.239	0.116	0.838	0.0375	0.0388	0.128
	Total Dose	214	6.76	95.4	1.25	0.906	1.98	3.28	4.36	2.03	1.12	1.22	6.82
135°	Neutron Dose	214	2.42	124	0.453	0.305	0.772	1.78	2.87	0.351	0.431	0.485	4.49
	Photon Dose	1.29	0.483	1.89	0.0506	0.00765	0.167	0.180	0.0856	0.755	0.0131	0.0142	0.107
	Total Dose	215	2.90	126	0.504	0.313	0.939	1.96	2.96	1.11	0.445	0.499	4.60
180°	Neutron Dose	213	8.10	154	1.32	0.884	2.65	4.55	5.82	1.11	1.27	1.43	10.6
	Photon Dose	1.27	0.788	2.23	0.140	0.0219	0.336	0.361	0.137	0.859	0.0315	0.034	0.213
	Total Dose	215	8.89	157	1.46	0.906	2.98	4.91	5.96	1.97	1.30	1.46	10.8

Value in table is multiplied by 1×10^{-17} Sv per source neutron

Dose Equivalent to ICRU Sphere

		Geometry											
		0	1	2	3	4	5	6	7	8	9	10	11
0°	Neutron Dose	213	275	294	281	281	281	282	277	337	279	281	272
	Photon Dose	1.27	3.24	3.79	4.10	3.62	4.06	4.05	3.50	4.95	3.01	3.00	3.47
	Total Dose	215	278	298	285	284	285	286	281	342	282	284	275
45°	Neutron Dose	213	16.9	109	11.0	9.89	14.9	20.6	13.6	20.7	10.5	11.0	40.2
	Photon Dose	1.28	0.934	1.47	0.532	0.270	0.701	0.826	0.54	1.01	0.309	0.310	0.902
	Total Dose	214	17.9	111	11.5	10.2	15.6	21.4	14.1	21.7	10.9	11.3	41.1
90°	Neutron Dose	212	6.10	94.1	1.14	0.869	1.73	3.04	4.25	1.19	1.08	1.18	6.70
	Photon Dose	1.28	0.663	1.33	0.116	0.0372	0.246	0.239	0.116	0.838	0.0375	0.0388	0.128
	Total Dose	214	6.76	95.4	1.25	0.906	1.98	3.28	4.36	2.03	1.12	1.22	6.82
135°	Neutron Dose	214	2.42	124	0.453	0.305	0.772	1.78	2.87	0.351	0.431	0.485	4.49
	Photon Dose	1.29	0.483	1.89	0.0506	0.00765	0.167	0.180	0.0856	0.755	0.0131	0.0142	0.107
	Total Dose	215	2.90	126	0.504	0.313	0.939	1.96	2.96	1.11	0.445	0.499	4.60
180°	Neutron Dose	213	8.10	154	1.32	0.884	2.65	4.55	5.82	1.11	1.27	1.43	10.6
	Photon Dose	1.27	0.788	2.23	0.140	0.0219	0.336	0.361	0.137	0.859	0.0315	0.034	0.213
	Total Dose	215	8.89	157	1.46	0.906	2.98	4.91	5.96	1.97	1.30	1.46	10.8

Value in table is multiplied by 1×10^{-17} Sv per source neutron

Dose Equivalent to ICRU Sphere

		Geometry											
		0	1	2	3	4	5	6	7	8	9	10	11
	Neutron Dose	213	275	294	281	281	281	282	277	337	279	281	272
0°	Photon Dose	1.27	3.24	3.79	4.10	3.62	4.06	4.05	3.50	4.95	3.01	3.00	3.47
	Total Dose	215	278	298	285	284	285	286	281	342	282	284	275
	Neutron Dose	213	16.9	109	11.0	9.89	14.9	20.6	13.6	20.7	10.5	11.0	40.2
45°	Photon Dose	1.28	0.934	1.47	0.532	0.270	0.701	0.826	0.54	1.01	0.309	0.310	0.902
	Total Dose	214	17.9	111	11.5	10.2	15.6	21.4	14.1	21.7	10.9	11.3	41.1
	Neutron Dose	212	6.10	94.1	1.14	0.869	1.73	3.04	4.25	1.19	1.08	1.18	6.70
90°	Photon Dose	1.28	0.663	1.33	0.116	0.0372	0.246	0.239	0.116	0.838	0.0375	0.0388	0.128
	Total Dose	214	6.76	95.4	1.25	0.906	1.98	3.28	4.36	2.03	1.12	1.22	6.82
	Neutron Dose	214	2.42	124	0.453	0.305	0.772	1.78	2.87	0.351	0.431	0.485	4.49
135°	Photon Dose	1.29	0.483	1.89	0.0506	0.00765	0.167	0.180	0.0856	0.755	0.0131	0.0142	0.107
	Total Dose	215	2.90	126	0.504	0.313	0.939	1.96	2.96	1.11	0.445	0.499	4.60
	Neutron Dose	213	8.10	154	1.32	0.884	2.65	4.55	5.82	1.11	1.27	1.43	10.6
180°	Photon Dose	1.27	0.788	2.23	0.140	0.0219	0.336	0.361	0.137	0.859	0.0315	0.034	0.213
	Total Dose	215	8.89	157	1.46	0.906	2.98	4.91	5.96	1.97	1.30	1.46	10.8

Value in table is multiplied by 1×10^{-17} Sv per source neutron

Dose Equivalent to ICRU Sphere

		Geometry											
		0	1	2	3	4	5	6	7	8	9	10	11
	Neutron Dose	213	275	294	281	281	281	282	277	337	279	281	272
0°	Photon Dose	1.27	3.24	3.79	4.10	3.62	4.06	4.05	3.50	4.95	3.01	3.00	3.47
	Total Dose	215	278	298	285	284	285	286	281	342	282	284	275
	Neutron Dose	213	16.9	109	11.0	9.89	14.9	20.6	13.6	20.7	10.5	11.0	40.2
45°	Photon Dose	1.28	0.934	1.47	0.532	0.270	0.701	0.826	0.54	1.01	0.309	0.310	0.902
	Total Dose	214	17.9	111	11.5	10.2	15.6	21.4	14.1	21.7	10.9	11.3	41.1
	Neutron Dose	212	6.10	94.1	1.14	0.869	1.73	3.04	4.25	1.19	1.08	1.18	6.70
90°	Photon Dose	1.28	0.663	1.33	0.116	0.0372	0.246	0.239	0.116	0.838	0.0375	0.0388	0.128
	Total Dose	214	6.76	95.4	1.25	0.906	1.98	3.28	4.36	2.03	1.12	1.22	6.82
	Neutron Dose	214	2.42	124	0.453	0.305	0.772	1.78	2.87	0.351	0.431	0.485	4.49
135°	Photon Dose	1.29	0.483	1.89	0.0506	0.00765	0.167	0.180	0.0856	0.755	0.0131	0.0142	0.107
	Total Dose	215	2.90	126	0.504	0.313	0.939	1.96	2.96	1.11	0.445	0.499	4.60
	Neutron Dose	213	8.10	154	1.32	0.884	2.65	4.55	5.82	1.11	1.27	1.43	10.6
180°	Photon Dose	1.27	0.788	2.23	0.140	0.0219	0.336	0.361	0.137	0.859	0.0315	0.034	0.213
	Total Dose	215	8.89	157	1.46	0.906	2.98	4.91	5.96	1.97	1.30	1.46	10.8

Value in table is multiplied by 1×10^{-17} Sv per source neutron

Dose Equivalent to ICRU Sphere

		Geometry											
		0	1	2	3	4	5	6	7	8	9	10	11
	Neutron Dose	213	275	294	281	281	281	282	277	337	279	281	272
0°	Photon Dose	1.27	3.24	3.79	4.10	3.62	4.06	4.05	3.50	4.95	3.01	3.00	3.47
	Total Dose	215	278	298	285	284	285	286	281	342	282	284	275
	Neutron Dose	213	16.9	109	11.0	9.89	14.9	20.6	13.6	20.7	10.5	11.0	40.2
45°	Photon Dose	1.28	0.934	1.47	0.532	0.270	0.701	0.826	0.54	1.01	0.309	0.310	0.902
	Total Dose	214	17.9	111	11.5	10.2	15.6	21.4	14.1	21.7	10.9	11.3	41.1
	Neutron Dose	212	6.10	94.1	1.14	0.869	1.73	3.04	4.25	1.19	1.08	1.18	6.70
90°	Photon Dose	1.28	0.663	1.33	0.116	0.0372	0.246	0.239	0.116	0.838	0.0375	0.0388	0.128
	Total Dose	214	6.76	95.4	1.25	0.906	1.98	3.28	4.36	2.03	1.12	1.22	6.82
	Neutron Dose	214	2.42	124	0.453	0.305	0.772	1.78	2.87	0.351	0.431	0.485	4.49
135°	Photon Dose	1.29	0.483	1.89	0.0506	0.00765	0.167	0.180	0.0856	0.755	0.0131	0.0142	0.107
	Total Dose	215	2.90	126	0.504	0.313	0.939	1.96	2.96	1.11	0.445	0.499	4.60
	Neutron Dose	213	8.10	154	1.32	0.884	2.65	4.55	5.82	1.11	1.27	1.43	10.6
180°	Photon Dose	1.27	0.788	2.23	0.140	0.0219	0.336	0.361	0.137	0.859	0.0315	0.034	0.213
	Total Dose	215	8.89	157	1.46	0.906	2.98	4.91	5.96	1.97	1.30	1.46	10.8

Value in table is multiplied by 1×10^{-17} Sv per source neutron

Conclusions

- The energy spectra through the sphere at 0° for a shielded source were similar to that of an unshielded source
- Multiple layers of steel and polyethylene decreased the unshielded dose by nearly a factor of 1,000
- At the 45° sphere, the addition of Boron-10 to the polyethylene reduced the number of neutrons below 1 MeV by 50%
- Depending on the application, the mass and cost of the shielding can be scaled back at the expense of increased dose

Future work

- Include additional objects around shielding to simulate environment
- Run scaled experiments in lab with 14.1 MeV neutron source to test calculations
- Study activation of steel when layered with polyethylene
- Build shielding

Thank you

Any questions?

References

1. Hussein E M A. Handbook on radiation probing, gauging, imaging and analysis, Volume II: applications and design. Norrwell, MA: Kulwer Academic Publishers; 2003.
2. International Commission on Radiation Units and Measurements. Measurement of dose equivalents from external photon and electron radiations. Bethesda, MD: Report 47; 1992.
3. International Commission on Radiation Protection. Data for use in protection against external radiation. Oxford: Pergamon Press; ICRP Publication 51; 1987.
4. Maruyama T, Bouts C J. Attenuation of 15 MeV neutrons in multilayer shields composed of steel, polyethylene and borated materials. Phys Med Biol 17: 420 – 424; 1972.
5. Shleien B, Slaback LA, Birky BK. Handbook of health physics and radiological health. 3rd ed. Baltimore, MD: Lipcott Williams and Wilkins; 1998.