MORSE Monte Carlo Shielding Calculations For The Zirconium Hydride Reference Reactor

C. E. Burgart
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Verification of DOT-SPACETRAN transport calculations of a lithium-hydride and tungsten shield for a SNAP reactor has been performed using the MORSE Monte Carlo code. Transport of both neutrons and gamma rays was considered. Importance sampling was utilized in the MORSE calculations. Several quantities internal to the shield, as well as dose at several points outside of the configuration, were in satisfactory agreement with the DOT calculations of the same.

A requirement exists to provide a low radiation environment for the crew of a space station with electrical power provided by a SNAP reactor. Therefore, a shield must be designed to provide adequate protection for the crew while having the smallest possible mass.

A shadow shield consisting of lithium hydride and tungsten has been designed and optimized for a 1-MW SNAP reactor using the ASOP optimization code (ref. 1). ASOP utilizes ANISN, a well-known one-dimensional discrete ordinates code (ref. 2), in performing the optimization. Another paper presented at this symposium discusses the optimization calculations in more depth (ref. 3). Subsequent to the ASOP optimization of the shield, calculations have been performed using DOT, a two-dimensional discrete ordinates code (ref. 4), and SPACETRAN, a transport code (ref. 5) for void regions, to verify that the dose constraints were met in the ASOP shield design. The dose constraint at the bottom of the shield 100 ft from the center of the core of the reactor was 10 mrem/hr. The dose constraint 100 ft from the center of the core at the side and top of the configuration was 100 rem/hr.

Verification of the DOT-SPACETRAN calculations has been performed using the MORSE Monte Carlo code (ref. 6). This code was chosen for several reasons. First was its ability to perform transport calculations in geometrically complex configurations in two and three dimensions. Second, since this problem includes transport of both primary neutrons and secondary gamma rays, MORSE's capability of performing the coupled calculation in a single job step on the computer is quite convenient. Furthermore, the ability to use the identical multigroup cross sections makes possible the checking of the transport calculations independent of the cross sections. Also, if further investigation is required, MORSE calculations can utilize considerably more cross-section groups with little sacrifice in calculation time.

The configuration considered in these calculations is shown in figure 1. This is a figure of revolution about an axis along the lower edge of the figure. (Note that plus Z and 0° is to the left, minus Z and 180° toward the crew is to the right.) The details in the core region include: region 1, the ^{235}U and zirconium-hydride core; regions 2, 3, and 4, the reflectors and poison for the control of the reactor; region 5, the stainless steel vessel; regions 8, the upper and lower stainless steel grid plates; and regions 9, the upper and lower sodium plenums. The shield is made up of lithium hydride in region 7 and tungsten in region 6.

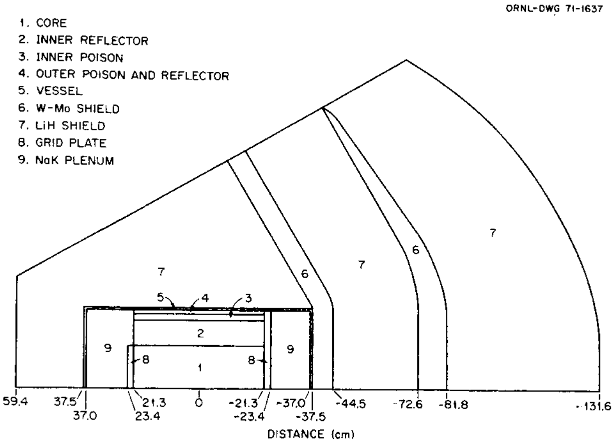

FIGURE 1.-Zirconium Hydride Reference Reactor Optimized 30° Shield.

The primary source of radiation for this problem was in region 1, the core of the reactor. A fission neutron source was distributed over the volume of the reactor according to previous DOT core calculations. The MORSE calculation was a fixed source calculation, i.e., multiplication of neutrons by fission was not allowed. However, the production of fission gamma rays was allowed.

These calculations were performed in 21 neutron groups and 18 gamma-ray groups with a P_3 expansion of the angular distribution. The DOT calculations were performed with 70 discrete angles and 4500 spatial mesh points. The MORSE calculations used the identical cross sections and the identical geometry, except for the capability of MORSE to smooth the stepped surfaces of the shield. The energy group structure, source neutron energy spectrum, and the neutron and gamma-ray flux-to-dose conversion factors are given in figures 2 and 3.

ORNL-DWG 71-1824

Neutron Group	Upper Energy (eV)	Source Spectrum (neutrons/sec)	Flux-to-Dose Conversion Factors [(mrem/hr)/neut/cm^2/sec]
1	14.918+6	1.641+7	1.50-1
2	10.000+6	3.115+8	1.50-1
3	6.7032+6	2.095+9	1.37-1
4	4.4933+6	6.948+9	1.32-1
5	3.0119+6	1.428+10	1.31-1
6	2.0190+6	2.139+10	1.25-1
7	1.3534+6	2.595+10	1.16-1
8	9.0718+5	2.763+10	1.06-1
9	5.5023+5	2.620+10	7.57-2
10	3.3373+5	2.296+10	5.51-2
11	2.0242+5	1.918+10	4.01-2
12	1.2277+5	1.422+10	2.45-2
13	4.0867+4	7.654+9	8.50-3
14	1.1709+4	0.0	5.00-3
15	3.3546+3	0.0	5.00-3
16	7.4852+2	0.0	5.00-3
17	1.6702+2	0.0	5.00-3
18	3.7266+1	0.0	5.00-3
19	8.3153+0	0.0	5.00-3
20	1.8554+0	0.0	5.00-3
21	4.1399-1	0.0	3.75-3
	2.50-2		

FIGURE 2.-Neutron Group Structure, Source Spectrum and Flux-to-Dose Conversion Factors.

ORNL-DWG 71-1825

Gamma Group	Upper Energy (MeV)	Flux-to-Dose Conversion Factors [(mrem/hr)/gr/cm^2/sec]
1	10.0	9.80-3
2	8.0	8.50-3
3	7.0	7.60-3
4	6.0	6.70-3
5	5.0	5.80-3
6	4.0	5.00-3
7	3.5	4.50-3
8	3.0	4.00-3
9	2.5	3.50-3
10	2.0	3.00-3
11	1.6	2.40-3
12	1.2	2.00-3
13	0.9	1.50-3
14	0.6	1.05-3
15	0.4	6.00-4
16	0.21	2.80-4
17	0.12	1.40-4
18	0.07	4.00-4
	0.01	

FIGURE 3.-Gamma-Ray Group Structure and Flux-to-Dose Conversion Factors.

Several quantities internal to the shield were calculated as a check on the source description. One of these quantities consisted of the average neutron and gamma-ray flux in all regions inside the stainless steel vessel. The average particle flux was calculated with MORSE utilizing a collision density estimator. Thus, the estimates far from the core or for small volumes have larger statistics. Figure 4 shows the results of this comparison between DOT and MORSE. The fractional standard deviations of the MORSE results are estimated to vary from 5% to 35%. Results are shown for both average neutron and gamma-ray fluxes. The agreement is quite good with the occasional discrepancies being attributed to the statistics involved in the MORSE calculations. The neutron results for the lower grid and lower plenum were obtained using source position biasing. Subsequently, the agreement is quite good with the statistics on the MORSE results being quite small.

ORNL-DWG 71-1823

COMPARISON OF AVERAGE PARTICLES FLUXES

Region	DOT Average Flux (particles/cm^2/sec)	MORSE Average Flux (particles/cm^2/sec)
1 (Core)		
neutrons/	5.39+13 [a]	5.50+13
gamma rays	9.43+13	9.39+13
2 (Inner Reflector)		
neutrons/	2.15+13	1.99+13
gamma rays	2.86+13	3.00+13
3 (Inner Poison)		
neutrons/	5.95+12	5.79+12
gamma rays	5.39+12	5.00+12
4 (Outer Reflector and Poison)		
neutrons/	3.47+12	3.64+12
gamma rays	2.06+12	1.94+12
8 (Upper Grid)		
neutrons/	1.36+13	2.06+13
gamma rays	1.87+13	2.36+13
8 (Lower Grid)		
neutrons/	7.83+12	7.80+12
gamma rays	9.18+12	8.31+12
9 (Lower Plenum)		
neutrons/	4.74+12	4.72+12
gamma rays	4.74+12	5.24+12

[a]Read as 5.39 x 10^{13}.

FIGURE 4.-Comparison of Average Particle Fluxes in the Core Region.

Another quantity considered was the average particle current leaving the core region through the top, side and bottom of the stainless steel vessel. Figure 5 shows the results of this comparison. The numbers in parentheses indicate the fractional standard deviations. The agreement again is quite satisfactory with all quantities lying within the statistics of the Monte Carlo calculation. The average neutron current leaving the bottom of the core region was obtained with source position biasing; thus, the statistics are quite small. This source position biasing was performed so that a leakage energy spectrum could be calculated with meaningful statistics in a reasonable time. This average neutron current spectrum through the bottom of the core region is shown in figure 6. The energy range shown is from 0.1 MeV to 10 MeV. The solid histogram is the result of a DOT calculation. The dashed histogram with error bars is the result of MORSE calculations. Below 370 keV, meaningful statistics were not obtained; thus, this comparison is only for fast neutrons. This comparison verifies the neutron transport in this energy range through the core and sodium plenum as it effects the transport in the shield. Any subsequent discrepancy between the MORSE and DOT results must be attributed to subsequent radiation transport between this surface and the detector. Also in future calculations it may be advantageous to use the DOT leakage current from the core as the source, and this comparison provides an added checkpoint.

ORNL-DWG 71-1826

COMPARISON OF AVERAGE PARTICLE CURRENTS

Surface	DOT Average Current (particles/cm^2/sec)	MORSE Average Current (particles/cm^2/sec)
Top		
neutrons/	0.96+12	1.02+12(.11) [a]
gamma rays	1.39+12	1.61+12(.35)
Bottom		
neutrons/	-7.01+11	-7.12+11(.03)
gamma rays	-13.6+11	-14.4+11(.21)
Side		
neutrons/	8.54+11	8.23+11(.04)
gamma rays	8.07+11	7.05+11(.18)

[a]Read as 1.02 x 10^{12} $\pm$ 11%.

FIGURE 5.-Comparison of Average Particle Currents Leaving the Core Region.

The dose at a point 100 ft from the top of the core and 100 ft from the side of the core was obtained with MORSE with a minimum of importance sampling. The dose 100 ft from the center of the core at the bottom of the configuration was a very deep penetration problem and thus required much importance sampling. Without this importance sampling the calculations of the dose at points outside the shield would have been impossible to perform. The overall attenuation of neutron dose from the center of the core to the edge of the shield was approximately 3 x 10^9. The attenuation through the shield alone was 2 x 10^8. The attenuation of gamma-ray dose from the center of the core to the edge of the shield was approximately 6 x 10^7. The attenuation through the shield alone was 2 x 10^6.

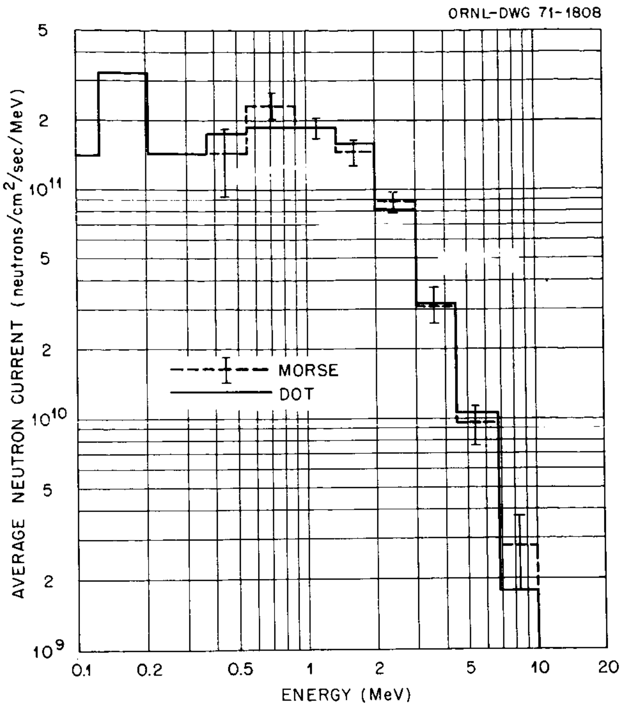

FIGURE 6.-Average Neutron Current Spectrum Through the Bottom of the Core Region.

The parameters used in the importance sampling were obtained from adjoint ANISN calculations. Also, the forward DOT calculations provided initial Russian roulette standards. The adjoint ANISN calculation was performed in slab geometry with the materials shown along the minus-Z axis of the shield; that is, core, stainless steel grid plate, sodium plenum, tungsten, lithium hydride, tungsten, and again lithium hydride. The source for the adjoint ANISN calculation was the detector response at the outermost surface of the slab. In the MORSE calculation, neutron source energy importance sampling was performed using the scalar importance calculated by ANISN in the core region. The neutron source position was biased according to the adjoint ANISN calculation. Source neutron and source secondary gamma-ray angle biasing was also performed. The importance was allowed to vary linearly from 0.01 in the backward direction to 0.99 in the forwardmost direction, that is, toward the bottom of the shield. The most important importance sampling was performed in the selection of particle-flight-path lengths. A form of the exponential transform was utilized. That is, the importance was assumed to vary exponentially along a flight path, i.e., $\exp(\beta\Sigma_t \ell)$. For the calculation of the dose at the bottom of the shield, the direction of greatest importance was assumed to be toward -z for z > 0 and away from a point at z = 50 for z < 0. The slopes of the exponentials, β, for each group and region were obtained by least-squares fitting the ANISN angular flux in the forwardmost direction with exponentials for a fixed number of mean-free paths. This was performed at each mesh point and then averaged over larger regions. The results of these fits may be expressed in terms of stretching parameters. These parameters can yield a good deal of insight into the particle transport in the shield. For instance, when the total response was used as a source for the adjoint ANISN calculation, path-length stretching parameters for fast neutrons were on the order of from 2 to 3 for the core first tungsten and lithium hydride regions, and on the order of 1-1/2 and less for the second tungsten region, whereas when the neutron response only was used as the source for the adjoint ANISN calculation path-length stretching parameters on the order of from 4 to 5 were calculated for all regions until the outer edge of the shield. This indicates the relative importance of neutrons to generating secondary gamma rays versus the importance of neutrons in contributing to the dose at the detector point.

Two other forms of importance sampling were also used. First, estimates were made of the dose at the bottom of the shield for only a fraction of the particle collisions occurring on the core side of the first tungsten region. When an estimate was made the weight was adjusted accordingly. Also, Russian roulette parameters were determined by observation of the number and weight of real collisions occurring in each group and in each region.

Estimation of the dose at the points outside the shield was performed utilizing a next-flight estimator. The results of these MORSE calculations and the DOT-SPACETRAN calculations are given in figure 7. The dose, 100 ft from the center of the core in mrem/hr, is given at several positions around the shield. The top dose, being the easiest quantity to calculate, yielded the best results. The DOT-SPACETRAN results fell within the statistics of the MORSE calculations. The dose falls off slightly in proceeding from 0° to 30° due to the increased thickness of the shield. At the side of the configuration, the reflectors on the reactor and the poison control drums, as well as the increased thickness of lithium hydride, contribute in making the dose somewhat less. At 90° the first tungsten region begins to have an effect. The neutron dose appears to be reasonable, considering the statistics. The gamma-ray dose calculated by DOT falls off a factor of 2 between 85° and 95°. The MORSE results at 90° are closer to the DOT results at 95°. The dose at the bottom of the shield should be relatively flat from 150° to 180° due to the shadow shield's nature of the configuration. The MORSE neutron dose agrees quite well with that calculated by DOT. The gamma-ray dose calculated by MORSE, however, appears to be somewhat low. Two possible reasons may be: (1) the MORSE results are not yet converged to the answer, or (2) DOT results are somewhat high. The latter may be caused by some larger spatial mesh intervals in the tungsten region of the shield. In any case, the agreement appears to be quite satisfactory. After all, the attenuation is 10^6 to 10^8, or more.

ORNL-DWG 71-1917

DOSE 100 FT FROM THE CENTER OF THE CORE
(MREM/HR)

	Angle	DOT Neutron Dose	DOT Gamma Dose	DOT Total Dose	MORSE Neutron Dose	MORSE Gamma Dose	MORSE Total Dose
Top	0°				1.64+5(.10)[a]	1.77+5(.16)	3.41+5(.10)
	5°	1.77+5	1.75+5	3.52+5			
	10°	1.78+5	1.77+5	3.55+5			
	15°	1.75+5	1.74+5	3.49+5			
	20°	1.45+5	1.35+5	2.79+5			
	25°	1.33+5	1.47+5	2.80+5			
	30°	1.28+5	1.40+5	2.68+5			
	80°	4.54+4	7.61+4	1.22+5			
	85°	3.39+4	7.52+4	1.09+5			
Side	90°				4.60+4(.35)	3.37+4(.15)	7.97+4(.21)
	95°	1.85+4	3.51+4	5.36+4			
	100°	6.65+3	7.23+3	1.39+4			
	165°	1.27	10.9	12.2			
	170°	1.27	10.7	12.0			
	175°	1.23	10.4	11.6			
Bottom	180°				1.35(.35)	5.90(.25)	7.25(.21)

[a]Read as $1.64 \times 10^5 \pm 10\%$.

FIGURE 7.-Comparison of the Dose of Points 100 Feet From the Center of the Core.

REFERENCES

1. ENGLE, WARD W., JR.: A Users Manual for ASOP, ANISN Shield Optimization Program. USAEC Report CTC-INF-941, Union Carbide Corporation, 1969.
2. ENGLE, W. W., JR.: A Users Manual for ANISN. USAEC Report K-1693, Union Carbide Corporation, 1967.
3. ENGLE, W. W., JR.: The Design of Asymmetric 4π Shields for Space Reactors. This paper presented in this NASA document.
4. MYNATT, F. R.; MUCKENTHALER, F. J.; and STEVENS, P. N.: Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding. USAEC Report CTC-INF-952, Union Carbide Corporation, 1969.
5. SOLOMITO, M.: SPACETRAN - A Code to Calculate Dose at Detectors at Various Distances From the Surface of a Cylinder. USAEC Report ORNL-TM-2592, Oak Ridge National Laboratory, 1969.
6. STRAKER, E. A.; STEVENS, P. N.; IRVING, D. C.; and CAIN, V. R.: The MORSE Code - A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code. USAEC Report ORNL-4585, Oak Ridge National Laboratory, 1970.